Shippingport Operations

FROM POWER OPERATION AFTER FIRST REFUELING TO SECOND REFUELING

MAY 6, 1960 TO AUGUST 16, 1961

Prepared by

DUQUESNE LIGHT COMPANY

and

**BETTIS ATOMIC POWER LABORATORY,
WESTINGHOUSE ELECTRIC CORPORATION**

Under Contracts with the

UNITED STATES ATOMIC ENERGY COMMISSION

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FOREWORD

This report describes operation of the Shippingport Atomic Power Station with the second seed of the first core. Similar reports have been issued summarizing operation with the first seed and refueling of Seed 1 with Seed 2.

The primary purpose of the Shippingport Project is to advance the basic technology of water cooled reactors through the design, development, testing and operation of a large power reactor as part of a utility system. A great deal of instrumentation, flexibility of operation and other special features have been incorporated into the design and construction of the reactor plant to facilitate the acquisition of test data and operating experience.

Operation of Shippingport has resulted in information on water reactor technology essential to the ultimate achievement of economic nuclear power. This information, based on both experimental and analytical results, includes work on the heat transfer, hydraulic characteristics and reactor physics of water reactors, in addition to development of experience with primary system components such as heat exchangers, reactor vessels, pressurizers, canned rotor main coolant pumps, control rod drive mechanisms and valves. We hope that through reports such as this, industry will benefit from the technological information and practical experience developed in the construction, testing and operation of Shippingport.

H. G. Rickover
United States Atomic Energy Commission

Philip A. Fleger
Chairman of the Board and President
Duquesne Light Company, Pittsburgh, Pa.

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INTRODUCTION

Development and design of the Shippingport Atomic Power Station was initiated in the spring of 1953, construction started in March of 1955, and the operations and test program began in December of 1957.

On October 9, 1959, the "seed", or enriched uranium component of Shippingport Core I was determined to be ready for replacement. This marked the end of the first extended operating and test period - one of sufficient duration to be productive of a significant body of experience and data. This experience and data were made available in DLCS-364 "Shippingport Operations from Start-up to First Refueling," which provides a summary of the pertinent information developed during that period.

The replacement of the first enriched uranium loading began on November 2, 1959 and proceeded concurrently with plant modification and maintenance. Refueling was completed and initial criticality was attained with Seed 2 on April 12, 1960. Power operation commenced on May 6, 1960 and full power was attained on May 7, 1960. Operation continued at load factors considerably higher than those in Seed 1 until August 14, 1961, at which time the Station was shut down in preparation for the second refueling. Actual refueling operations commenced on August 16, 1961.

This report is concerned with the operation of the Shippingport Atomic Power Station during Seed 2 lifetime. This discussion deals primarily with experience with the nuclear portion of the station, with the "conventional" portion being introduced only as it relates to over-all station operation or to unusual problems which result from the use of conventional equipment with a pressurized water reactor.

Recognizing the wide range of interests directed toward the additional Shippingport experiences, it was concluded that this report could best be presented in five parts, in much the same manner as the original operations report. Part I consists of summary reviews of possible interest to management, and Parts II through V consist of detailed reports of interest to engineering and scientific personnel. Also included in Part I is a detailed Chronology which, either scanned or studied, provides an over-all view of what has occurred at Shippingport during the period.

It is assumed that the reader has more than a casual acquaintance with the background and design of the Shippingport Station. Those interested in detailed information on the design of the station are referred to the book, "The Shippingport Pressurized Water Reactor" published by the Addison-Wesley Publishing Company in 1958.

Consistent with the original premise that Shippingport was built to provide information and not power at competitive costs, and must, therefore, be regarded as a test facility, this report includes no analysis of kilowatt hour costs or cost reductions that might have been achieved had Shippingport been built and operated solely as a power producer. It should be noted that during the 15 months of Seed 2 operation, the plant was utilized to conduct numerous reactor plant and core tests at various stages of core depletion to improve the understanding of water cooled reactors and to train personnel in reactor plant operations. Despite the extensive low-power operation required by this testing and training, the 514,300,000 KWH gross generated during Seed 2 operation was equivalent to operating the plant at full power 70 percent of the time. During Seed 2 operation the Shippingport Station was used for testing, training, and power generation more than 97 percent of the time; the reactor plant was shut down for maintenance less than 3 percent of the time.

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PWR OPERATION HIGHLIGHTS

	SEED 1	SEED 2
Initial criticality for testing	December 2, 1957	April 12, 1960
Full power operation	December 23, 1957	May 6, 1960
Lifetime	5806 EFPH	7900 EFPH
Total electricity generated (gross output per seed)	388,500,000 KWHR	514,300,000 KWHR
Average load factor	37%	70%
Average load factor excluding testing	75%	97%
Date refueling started	November 2, 1959	August 16, 1961
Working days required to complete refueling and return to full power	156	63

Other reports on PWR Operation, Testing, Design, and Maintenance are listed in WAPD-PWR-1606 (Revised), The Shippingport Pressurized Water Reactor Project Catalog of Document Abstracts, December 1961 - This document lists over 1500 reports on PWR which are available through the Office of Technical Services, Department of Commerce, Washington 25, D.C.

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CONTENTS

PART I

GENERAL REVIEW OF STATION EXPERIENCE

CHAPTER 1

OPERATION AND TESTING

Introduction

The Core I Seed 2 operating period was devoted primarily to developing useful information on the long-term behavior of the reactor and major plant components. This includes the following:

1. Obtaining extensive information on the depletion capabilities of natural UO_2 blanket fuel elements.

2. Obtaining information on the physics characteristics and thermal performance of seed and blanket cores as a function of fuel depletion, and demonstrating the continual generation of a high fraction of the core power from the natural uranium blanket.

3. Acquisition of data on the performance of major plant systems and components, to obtain a realistic basis for judging the adequacy of component design specifications.

In order to obtain these objectives, the test schedule consisted basically of six full power runs, each approximately 1200 hours in length, separated by intervals of periodic reactor physics and equipment performance type testing. In addition, the dynamic operating characteristics of systems were investigated to determine their individual adequacy as well as their integrated performance characteristics.

The station was operated successfully both as a peak and base load unit in accordance with the test program, the student training program (refer to Part I, Chapter 2), and the power requirements of the Duquesne Light System. The reactor plant provided a stable heat source with a high degree of flexibility as evidenced by the numerous plant start-ups and shutdowns that were conducted during Seed 2 lifetime for test purposes. Experience to date indicates that this type of reactor power station could be incorporated readily into any electrical network system and operated as either a peak or base load station, depending upon economic considerations.

Operations

Seed 2 operated with a much higher station load factor (approximately 70% for Seed 2 compared to 37% for Seed 1) because of the absence of major equipment problems and fewer station shutdowns for test purposes. Figure I-1 depicts the accumulated gross generation incurred during Seed 2 life from start of power operation on May 6, 1960, to shutdown for Seed 2 - Seed 3 refueling on August 14, 1961. Table I-A contains the list of tests which were performed at the times specified by the numbers on Figure I-1.

During this fifteen-month period 7900 equivalent full power hours (EFPH) of operation were obtained. Full power capability for Seed 2 ended on June 10, 1961, after 7528 EFPH, when the reactivity of the core decreased to the point where the station could no longer be maintained at 60 MW net electrical output under normal reactor coolant temperature, pressure, and equilibrium xenon conditions. The station was then operated at reduced electrical output and reduced average coolant temperature until August 14, 1961, when it was shutdown for the installation of Seed 3.

Variations in generator load and reactor coolant temperature throughout Seed 2 life are presented in Part I, Chapter 3, Chronology. Subjects of particular interest relating to Seed 2 operation include the following items:

1. Blanket performance capabilities.

2. The station capability of operating as a peak and base load station.

3. Reactor stability.

4. Improved equipment reliability.

The natural uranium blanket performed satisfactorily throughout Seed 2 lifetime. Core instrumentation indicated that the percentage of core power produced by the blanket varied from 53% near the beginning of Seed 2 lifetime to 56% near the end of Seed 2 lifetime (refer to Part II). Only two blanket fuel elements (J-5 and K-8) were suspected of having defective tubes. These two fuel elements were removed at the Seed 2 - Seed 3 refueling and examined at the Idaho test facility. One tube in both elements were found to contain manufacturing defects of less than 5 mils diameter.

Station Capability

As exhibited during Seed 1 operation, the station again operated satisfactorily when subjected to various load swings and relatively fast start-ups and shutdowns. Load transient testing (DLCS 36301) conducted in July, 1960, and the sixty station start-up tests (DLCS 35002) performed in June, 1961, demonstrated the station's capability in this respect.

The longest uninterrupted base load operation (1357 EFPH) for Seed 2 occurred from April to June, 1961. It was found that Shippingport, as most conventional stations, accepts a base load operation very well and it is expected that longer extended power runs will be performed successfully.

Reactor Stability

The reactor was inherently stable throughout Seed 2 operation. As had been anticipated, late in seed life operator action in terms of rod motion was required during fast load reductions from full power in preparation for xenon transient testing because of the gradually decreasing effect of the negative temperature coefficient.

A radial flux oscillation deliberately induced into the core as part of a test revealed a damped radial oscillation power behavior and confirmed that Seed 2 was stable with respect to spontaneous radial oscillations. This test is discussed in greater detail in Part II, Chapter 3, Special Seed 2 Physics Tests.

Equipment Reliability

In general, the station equipment that presented major operating problems during Seed 1 operation was corrected by design modifications. These problems are discussed in detail in Part I, Chapter 6, Maintenance.

Three reactor coolant pumps operated without difficulty during Seed 2. The fourth reactor coolant pump, which had failed to start late in Seed 1 life, again exhibited this problem several times during initial Seed 2 testing. However, the pump did operate satisfactorily when it was not subjected to frequent cycling. Prior to the end of Seed 2 life this reactor coolant pump and volute were removed because of operating limitations and to prepare the loop to accept a pump for Core II. Subsequently, a pump identical in design to the other three reactor coolant pumps and a modified volute were installed for Seed 3 operation.

Steam generator leakage (primary to secondary) was detected on three of the four installed units. However, the worst leakage in the affected units was so small that it could be determined only by radiochemical methods based on iodine isotopes in the boiler water. Since this leakage presented no operating problem, all steam generators continued to operate satisfactorily during Seed 2 lifetime.

The turbine plant equipment developed two serious problems. Two of the three installed feedwater heaters developed major tube leakage and required retubing prior to power operation in Seed 3. The second problem occurred when tube failure due to chemical attack was encountered in the air ejector and necessitated complete retubing of one section of the unit. For specific details on all station equipment problems, refer to Part IV, Chapter 1.

Testing

The test information obtained during Seed 2 operation is generally comparable to that obtained during Seed 1 even though the cores differed in several respects. The over-all effects of increased fuel loading, the introduction of natural boron poison in Seed 2, and the changes in reactivity contribution of the blanket (natural uranium oxide) region are discussed in Part II, Reactor Physics Performance.

Approximately 375 performances of the 130 basic tests were completed during Seed 2. These tests are categorized into four groups, namely, Reactor Physics, Plant Performance, Radiation and Chemistry, and Equipment Performance type tests. Test evaluation reports are available through the office of Technical Services, Oak Ridge, Tenn. A brief discussion and a specific example of each of the four test groups is presented below.

Key numbers shown on the accumulated gross generation curve in Figure I-1 categorize the major tests according to definite operating periods. The tests are identified in Table I-A.

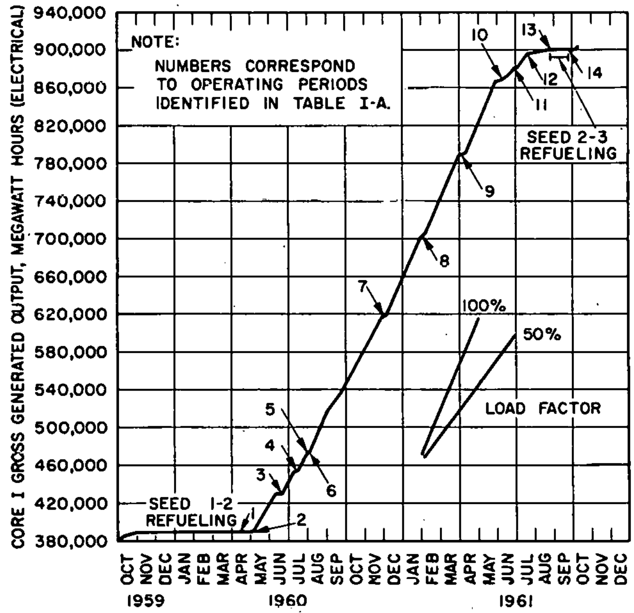

Figure I-1. Accumulated Gross Generation - Core I Seed 2.

TABLE 1-A

MAJOR TESTS PERFORMED DURING CORE I SEED 2

Key Number	Period	DLCS Number	Test Title
1	Initial approach to criticality for Core I, Seed 2 on April 12, 1960	16001	Initial Approach to Criticality
2	Initial Seed 2 Power Operation, May 6, 1960	35002	FEDAL System (Operation During Plant Startup)
		35601	Reactor Coolant Fission Product Activity
		36201	Reactivity Lifetime Test
2-3	May 7 to June 10, 1960 (0 to 761.6 EFPH)	15102	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
		34901	Station Performance of Steady State Loads
		35801	Reactor Pressure Drop and Coolant Flow Characteristics
		36201	Reactivity Lifetime Test
3	June 10 to 23, 1960	14801	Control Rod Drive Mechanism Periodic Test
		15601	Xenon Transient Tests
		36001	Flow Coastdown Test

TABLE 1-A (Cont'd)

MAJOR TESTS PERFORMED DURING CORE I SEED 2

Key Number	Period	DLCS Number	Test Title
4-5	June 23 to Aug. 13, 1960	15602	Xenon Transient Tests
		35002	FEDAL System (Operation During Plant Startup)
		35501	Steam Generator Test
		35502	Steam Generator Test
6	Aug. 13 to Aug. 19, 1960	15601	Xenon Transient Tests
		15603	Xenon Transient Tests
		34401	Rapid Station Shutdown
6-7	Aug. 19 to Nov. 25, 1960 (1648.9 to 3564.4 EFPH)	15102	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
		27601	Controlled Safety Test - Rod Withdrawal Transients
		36301	Dynamic Response of Reactor Plant to Load Swings
		37701	Effect of Steam Generator Performance on Core Power Distribution
7	Nov. 25 to Dec. 7, 1960	14901	Control Rod Position for Criticality
		14902	Control Rod Position for Criticality
		15101	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
8	Jan. 28 to Feb. 3, 1961 (4833.8 EFPH)	15102	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
		30501	Radiation Survey of the Reactor Vessel Head
9	March 30 to April 14, 1961 (6144.8 to 6171.3 EFPH)	14701	Determination of Reactor Coolant System Pressure Drop
		14901	Control Rod Position for Criticality
		14902	Control Rod Position for Criticality
		15101	Determination of Coefficients of Reactivity
		15102	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
		35002	FEDAL System (Operation During Plant Start-up)

TABLE 1-A (Cont'd)

MAJOR TESTS PERFORMED DURING CORE I SEED 2

Key Number	Period	DLCS Number	Test Title
9-10	April 14 to June 10, 1961 (6171.3 to 7528.3 EFPH)	15102	Determination of Coefficients of Reactivity
		36201	Reactivity Lifetime Test
		38801	Axial Flux Perturbation Test
10-11	June 10 to July 4, 1961	14801	Control Rod Drive Mechanism Periodic Test
		14901	Control Rod Position for Criticality
		14902	Control Rod Position for Criticality
		15003	Calibration and Intercomparison of Control Rods
		15101	Determination of Coefficients of Reactivity
		15601	Xenon Transient Tests
		30501	Radiation Survey of the Reactor Vessel Head
		35002	FEDAL System (Operation During Plant Start-up)
11-12	July 4 to July 16, 1961	36201	Reactivity Lifetime Test (75% Reactor Power)
		38802	Radial Flux Tilt Test
12-13	July 16 to August 14, 1961 (7900.7 EFPH)	35002	FEDAL System (Operation During Plant Start-up)
		36201	Reactivity Lifetime Test (50% Reactor Power)
13	August 16, 1961	---	Start of Seed 2-3 Refueling
14	Initial reactor criticality for Core I, Seed 3 on October 7, 1961	16001	Initial Approach to Criticality

Reactor Physics

Periodic reactor physics testing was conducted to provide information on the core operating characteristics as a function of core depletion. This type of information was necessary so that a comparison could be made between design and actual data. This comparison is then evaluated in order to improve analytical techniques used in developing future designs. Factors that are determined by this type of testing are temperature and pressure coefficients of reactivity, rod worth, core symmetry, temperature defect and excess shutdown reactivity. Comparison of predicted core performance with data at full power with equilibrium xenon and samarium were employed more extensively for Seed II, whereas during Seed I, comparisons were made using the low power, xenon free, test results. Physics tests, in general, confirmed results obtained during Seed 1. During Seed 2 lifetime, a significant increase in the blanket reactivities was observed. Measurements also indicated that there was a general decrease in the negative temperature coefficient from -2.3 to -0.8×10^{-4} /°F during Seed 2 life.

Example.

DLCS 15601--Xenon Transient Test.

The purpose of this test was to determine core reactivity as a function of existing xenon concentrations (equilibrium to maximum) throughout core life.

Immediately after a rapid shutdown from full power operation, the critical controlling group rod heights and worths were measured by observing approximately four reactor periods at each fixed group height during the xenon transient. As the core reactivity changed due to the transient, the rod group height was changed as required to maintain reactor criticality. The reactor period was obtained by observing the time required for the compensated ion chamber current to increase by the factor of "e."

The data from this test were used to determine the reactivity change between equilibrium and peak xenon conditions and thus obtain rod worth values.

Plant Performance

A number of plant performance type tests were performed at both steady-state and transient power conditions. The reactivity lifetime test (DLCS 36201) provided information on fuel depletion, the thermal performance of the core, and over-all plant performance when subjected to prolonged steady-state power operations. Relatively close agreement between analog simulation studies and actual load transient testing performed during Seed 1 decreased the need for periodic transient testing during Seed 2.

Example.

DLCS 36301 - Dynamic Response of Reactor Plant to Load Swings.

The purpose of this test was to determine the actual operating characteristics of the reactor plant systems, when integrated with the turbine plant systems, as the station was subjected to load swings.

Stable station operating conditions were established prior to each load transient. All load transients were performed with reactor control in manual in order to evaluate the individual effects of temperature coefficient, rod motion, and pressurizer spray operation. High-speed recorder equipment was utilized to record the major plant parameters that were affected by transient load conditions.

The parameter changes recorded during actual load swing testing were then compared to the results of analog simulation studies. Since no major plant changes have been made since original construction of the plant, and the temperature coefficients of Seed 1 and Seed 2 were relatively the same, all load transient tests conducted to date produced similar plant responses. These transient responses were satisfactory in all respects.

Radiation and Chemistry

Numerous tests of this type were conducted to determine the quantity and composition of crud, radiation level changes in various portions of the plant, location of hot spots, crud traps and other areas of preferential crud deposition as a function of plant operating conditions, and core power history. Another area of investigation involved tests to determine the possibility of cladding defects in blanket fuel elements.

Example.

DLCS 35002 - FEDAL System (Operation During Plant Start-up).

The purpose of this test was to determine if defective blanket fuel elements existed in the core locations previously found to have high levels of delayed neutron emitter activity.

During station start-up, the FEDAL system recorders were observed for any peaks in activity levels while monitoring sample flow from a specific blanket element. (A significant peak or "burst" of activity level is indicative of a defective element.)

All blanket fuel elements were monitored at least once during Seed 2 operational start-ups. Two elements, J-5 and K-8, exhibited symptoms of defects during the test performance. Both of these blanket assemblies were removed during Seed 2 - Seed 3 refueling and further investigation revealed small (less than 5 mil) defects.

Equipment Performance

Various types of tests were performed on installed systems and components in order to verify initial satisfactory performance, to investigate and evaluate equipment performance periodically, and to confirm the performance of modified equipment. Initial performance type tests were conducted for the newly installed reactor plant cooldown and temperature control system and the data acquisition system.

Example.

DLCS 12901 - Reactor Protection System

The purpose of this test was to verify the trip point settings and response time of the power-to-flow equipment.

The proper operation and response time of the reactor protection system circuitry was tested by the application of simulated input signals. Response times were obtained by evaluating high-speed recorder tracings obtained from specific stages of the circuitry tested. A comparison of the test data with the previously obtained results and the design values provided assurance that the system would operate satisfactorily during power conditions.

CHAPTER 2

NUCLEAR POWER STATION TRAINING PROGRAM

Introduction

In order to provide personnel from interested organizations with an opportunity to acquire a thorough practical background in nuclear power station operation and maintenance, a Nuclear Power Station Training Program complete with facilities and staff was established at the Shippingport Atomic Power Station. This program was designed to give supervisory personnel a practical background in the particular fields of science and engineering not normally associated with conventional fossil fuel power stations, but which are essential to safe and efficient operation of a nuclear power station. The course also provides an opportunity for actual experience in operation, maintenance, health physics, chemistry, and associated fields.

The program, facilities, and staff were established originally to provide college graduates with all the necessary qualifications and experience required for supervisory positions in nuclear plant operation and maintenance. Due to subsequent limitations, however, the degree of operational training initially planned could not be provided. The amount of practical operations training is limited by the Shippingport Test Program and its demand for continued reactor full power operation.

Prerequisites and Application

For admission to the course, students must be able to comprehend and speak the English language fluently, and must have graduated from an accredited high school. They should have the following academic material well in mind by virtue of recent academic courses, review, or self study prior to entering the Program:

1. Good basic understanding of electrical fundamentals.

2. Basic understanding of electronics.

3. Principles of high school physics. (Some knowledge of nuclear physics desirable)

4. High school algebra and trigonometry. (Calculus desirable)

5. High school or college chemistry.

6. Ability to read drawings, schematics, curves, and graphs.

In addition, domestic students will normally be expected to have had approximately six months of reactor or critical facility operating experience. When exceptions are to be made, each case will be judged on its individual merits.

Additional information on the Program may be obtained from the U. S. Atomic Energy Commission, Division of Reactor Development, Education and Training Branch, Washington 25, D. C.

Curriculum

Each four-month training period is divided into three phases—general classroom training, specialized classroom training, and practical training:

1. General Classroom Training - Four weeks of classroom time is devoted to general training in Systems, Components, and Operation; Chemistry, Radiochemistry, and Radiation Safety; Elements of PWR Physics and Control; and, Reactor Plant Instrumentation and Control. Each student, regardless of his specific interest, is required to take this training. Details in classroom training can be found in Reference 1.

2. Specialized Classroom Training - During this phase of training, the class is divided into groups according to specific interest (i. e. Operations, Chemistry and Health Physics, Instrumentation and Maintenance, or Test and Analysis). Each group meets separately with an instructor for a period of three weeks to cover specific details in their respective areas in preparation for the third phase of training.

In the case of operational students, this period is primarily devoted to a detailed study of those systems involving reactor operation and control. Typical subject material is as follows:

Lesson No.	Subject
1-2	Power Distribution
3-4	Primary Plant Instrumentation
5-6-7	Nuclear Instrumentation System
8-9-10	Rod Control System
11-12-13	Reactor Protection System
14	Plant Start-Up and Shutdown

3. Practical Training - The third phase of training consists of approximately eight weeks of practical experience in that phase of station operation in which each student is interested. Students interested in Maintenance, Instrumentation, and Test and Analysis work are assigned to the Shippingport Station maintenance, instrument, and test departments, respectively, to follow the work being done in these departments at the time. The daily curriculum for operating students is largely governed by plant operations and testing. The basic purpose is to give each student as much operating experience as possible while operating the plant at power. In general terms, the daily schedule includes:

   a. Two hours of classroom instruction in one of the following subjects:

      (1) Primary Plant Fluid Systems

      (2) Primary Plant Instrumentation System

(3) Primary Control System

(4) Primary Plant Systems Operating Procedures

(5) Casualty and Emergency Operation

(6) Nuclear Instrumentation System Alignment and Test Set Operation

(7) Pre-critical Check-off List (walk through check-off list)

(8) Review of those subjects covered during previous academic and practical training sessions.

b. One-hour station tour to supplement the lecture.

c. One hour at the reactor console for each student.

d. Four hours for study, individual discussions with instructor, or participation in any plant operation as directed by the Station Operating Engineer.

Two days each are spent in Chemistry, Radioactive Waste Disposal, and Health Physics to allow the operating students time to observe the operations of these departments.

In this third phase, the station is made available specifically for training. Each student will act as the reactor operator and perform all reactor and reactor console operations associated with the hot start-up and shutdown procedures during a full shift. At the beginning of each shift the station will be in a hot shutdown condition. The students will perform Part 2 of Pre-critical Check List S-R3 (lastest issue) and align two NIS channels. Prior to criticality, the students who are not operating the reactor will determine reactor xenon conditions and maximum permissible power levels, and perform those steps of the Pre-critical Check List which are necessary for safe station operations.

The student will then start-up the reactor and station. The station will then be loaded to approximately 35 MW gross electrical, held at load for 1/2 hour or more, and then shut down prior to the end of the shift to establish hot shutdown conditions for the next shift. If time permits, additional training in the form of control rod transfers and criticality attainment should be carried out during the shift.

Upon completion of these three phases of training, all students return for a one-week review prior to final written and oral examinations and graduation.

Training During Seed 2

Three groups of students, designated as Classes 3, 4 and 5, completed this training program during the period of Seed 2 operations. A summary of these classes in terms of areas of interest, sponsoring organizations, and dates is presented in Tables I-B, I-C, and I-D. The fact that certain sponsoring groups repeatedly send students to this program attests to its success.

Class 6

Inasmuch as the Seed 2 - Seed 3 refueling program was scheduled for the summer and fall of 1961, and the period between the end of the refueling program and the holiday season would not permit the completion of a class, Class No. 6 is planned for January 8, 1962. Thirteen students representing the Philadelphia Electric Company and the governments of the United Kingdom and Japan have been accepted for attendance. Present planning indicates that this class will be in session until May, 1962, and will be conducted in the same manner as Class Nos. 4 and 5.

Summary

Nine domestic organizations and eight foreign governments have thus far been represented, in the five Nuclear Power Station Training Program classes held during the period February 9, 1959, to May 17, 1961. Sixty-seven operations students, 11 chemistry and health physics students, and 2 maintenance students have completed the program.

TABLE I-B

COMPOSITION OF FIRST FIVE NUCLEAR POWER STATION TRAINING PROGRAM CLASSES

	No. 1	No. 2	No. 3	No. 4	No. 5	To Date
Starting date	Feb. 9, 1959	July 13, 1959	April 4, 1960	Aug. 8, 1960	Jan. 16, 1961	
Ending date	July 2, 1959	Oct. 30, 1959	July 28, 1960	Nov. 23, 1960	May 17, 1961	
Operations students	17	8	15	15	12	67
Chemistry students	1	2	1	1	2	7
Health Physics students	1	2	1	-	-	4
Maintenance students	1	-	1	-	-	2
Totals	20	12	18	16	14	80

TABLE I-C

DOMESTIC ORGANIZATIONS REPRESENTED IN THE NUCLEAR POWER STATION TRAINING PROGRAM

	No. 1	No. 2	No. 3	No. 4	No. 5	To Date
American Electric Power Service Corporation	1	-	-	-	-	1
Consolidated Edison Company of New York, Inc.	5	2	11	11	-	29
Northern States Power Company	1	1	-	-	-	2
Philadelphia Electric Company	2	1	-	-	-	3
Power Reactor Development Company	2	2	-	-	-	4
Rural Cooperative Power Association	1	-	-	-	-	1
Saxton Nuclear Experimental Corporation	-	-	1	-	-	1
Wentworth Institute	1	-	-	-	-	1
Yankee Atomic Electric Company	3	1	-	-	-	4
Totals	16	7	12	11	-	46

TABLE I-D

FOREIGN GOVERNMENTS AND ORGANIZATIONS REPRESENTED IN THE NUCLEAR POWER STATION TRAINING PROGRAM

	No. 1	No. 2	No. 3	No. 4	No. 5	To Date
Belgium						
Centre d'Etude de l'Energie Nucleaire	-	2	-	-	-	2
Formosa						
Taiwan Power Company	1	-	-	-	-	1
India						
Atomic Energy Establishment	1	-	-	-	-	1
Italy						
Centro Autonomo Militare Energia Nucleare	1	-	-	-	-	1
Societa Elettronucleare Nazionale	-	1	-	-	-	1
Japan						
Chubu Electric Power Company	-	-	1	-	-	1
Chugoku Electric Power Company	-	-	-	-	1	1
Kansai Electric Power Company	1	1	-	-	-	2
Kyushu Electric Power Company	-	1	-	-	1	2
Tokyo Electric Power Company	-	-	1	-	-	1
Sweden						
Swedish State Power Board	-	-	-	-	1	1
Switzerland						
Atomelectra, Ltd.	-	-	1	-	-	1
United Kingdom						
Rolls-Royce and Associates, Ltd.	-	-	3	4	6	13
Vickers-Armstrongs, Ltd.	-	-	-	1	5	6
Totals	4	5	6	5	14	34

REFERENCES

1. J. E. Gray, W. H. Hamilton, and W. E. Wynne, eds., "Shippingport Operations; From Start-up to First Refueling December 1957 to October 1959," DLCS-364.

CHAPTER 3

CHRONOLOGY

Introduction

The operation of the station during PWR Core I Seed 2 life is described in the day-by-day chronology which follows. All information has been taken from the station records and is presented in abbreviated form. Outstanding daily operation, maintenance, and testing activities are listed, along with the time and cause of every safety shutdown. Figure I-2 shows net generated output and reactor plant temperature for each month, the equivalent full power hours at key points, and the time and date of every safety shutdown (scram).

In this summary station output is often expressed in gross electrical megawatts as well as percent of reactor power. Since the gross electrical output is influenced by conditions such as boiler feedwater heater outages, climatic river temperature changes, and the number of reactor coolant loops in service, the term percent reactor power is more meaningful in describing reactor operation. In general, station output was varied to maintain the reactor at 100 percent power during Core I Seed 2 lifetime. Operating the reactor at 100 percent power is equivalent to 231 megawatts thermal output; normally this corresponds to 67 MW gross electrical at 60 MW net electrical output. The term equivalent full power hour (EFPH) is also used and is defined as operation of the reactor at 100 percent power for one hour or the equivalent thereof.

The period covered in this chronology is from reactor pre-critical preparations on April 12, 1960 to the plant cooldown for the start of Core I Seed 3 refueling operations on August 16, 1961.

APRIL, 1960

Apr. 12 — Performed initial approach to criticality test with the plant at 135°F. Reactor critical for first time on Core I Seed 2 at 8:06 P. M.

Apr. 13-18 — Station remained shut down for testing. 1A, 1C, and 1D loops were in service at 135°F and 450 psig with modified core removal cooling in service to maintain plant temperature for testing. 1B loop remained isolated for heat exchanger repairs. The following tests were performed during this period:

a) Calibration and intercomparison of control rods.
b) Operational investigation of nuclear instrumentation.
c) Control rod positions for criticality.
d) Reactor protection system operation.

Apr. 19 — Increased plant temperature to 200°F with 1A, 1C, and 1D reactor coolant pumps in slow speed for determination of coefficients of reactivity test.

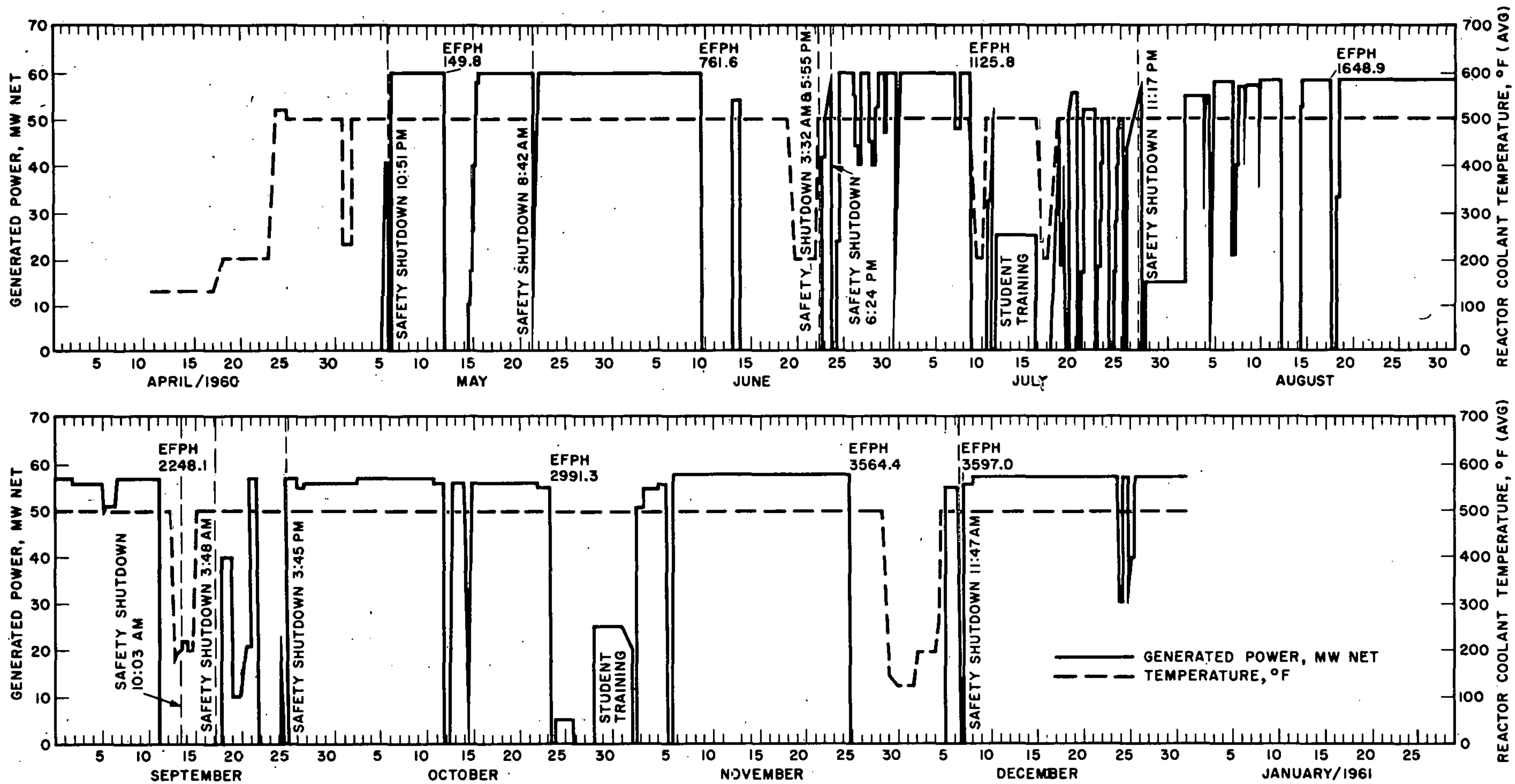

Figure I-2. Net Generated Output, Monthly Reactor Temperature, and Safety Shutdown Dates.

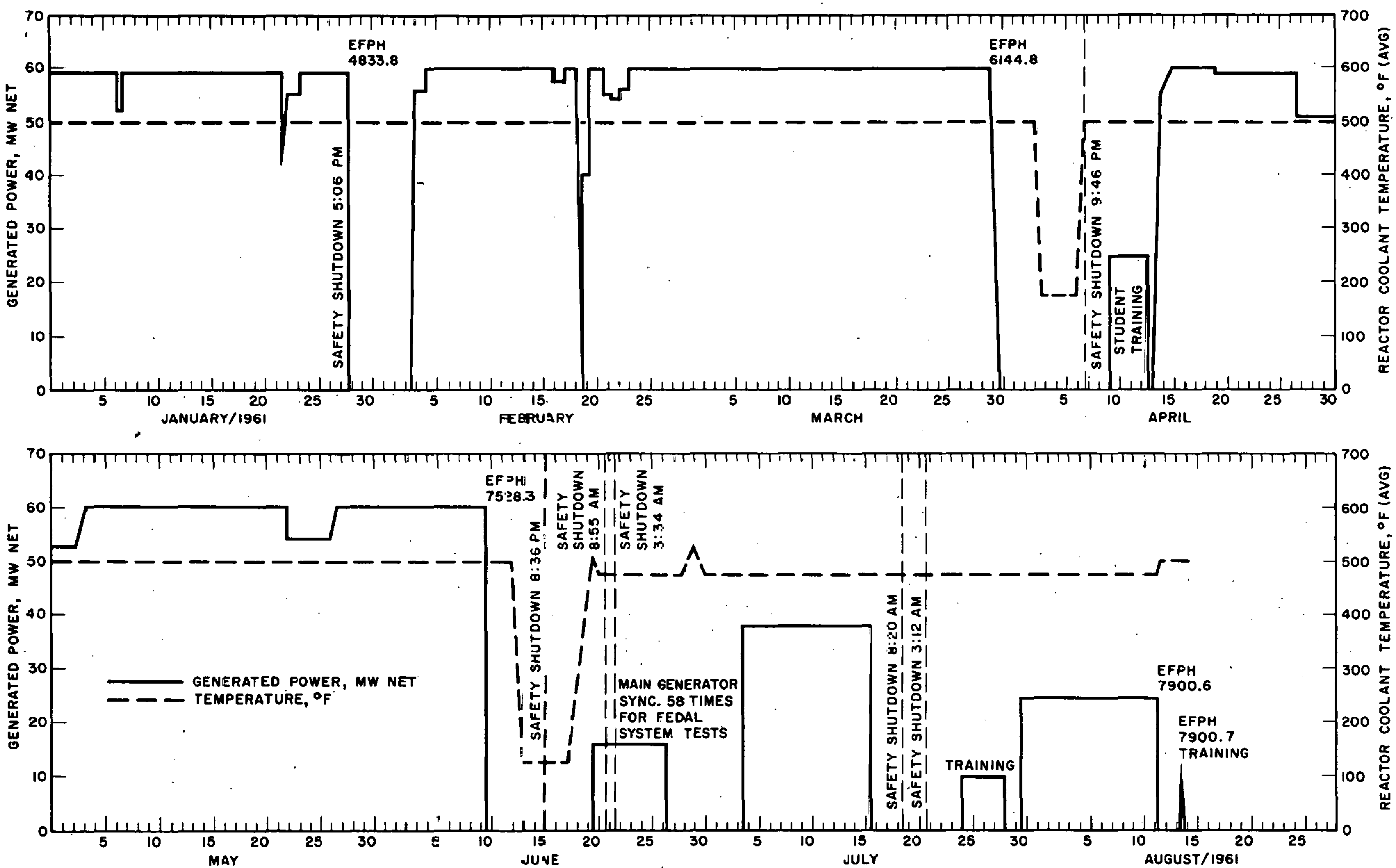

Figure I-2 (Cont.) Net Generated Output, Monthly Reactor Temperature, and Safety Shutdown Dates.

Apr. 20-21 — Performed reactor protection system response test and periodic calibration of pressure instrumentation test.

Apr. 22 — During a hydrostatic test of 1B loop at 2750 psig following repairs to 1B heat exchanger, a small leak was detected at the reactor coolant pump gasket. The loop was isolated to inspect the pump gasket.

Apr. 23-24 — Vented reactor vessel d/p cells; performed reactor plant container integrity test and 1D reactor coolant pump operational test.

Apr. 25 — Reactor taken critical for plant heat-up to 450°F. The 1A, 1C, and 1D reactor coolant pumps were then placed in fast speed to heat up plant to 520°F for determination of coefficients of reactivity test.

Apr. 26 — At 9:30 P.M. four safety insertions occurred due to a faulty log microammeter in the nuclear instrumentation system. It was necessary to replace the log microammeter to correct the condition.

Apr. 27-30 — Plant at 500°F and 1800 psig. Performed the following tests:

a) Control rod positions for criticality.
b) Operational investigation of nuclear instrumentation.
c) Periodic calibration of pressure instrumentation.
d) Intercalibration of temperature sensing elements.
e) Periodic reactor plant leak rate.
f) Control rod mechanisms precritical check.
g) Reactor protection system.
h) Reactor pressure drop and coolant flow characteristics.

Apr. 29 — 1D loop was isolated and cooled down for steam valve repairs.

<u>MAY, 1960</u>

May 1 — Plant at 500°F with 1A and 1C loops in service. Performed successful hydrostatic tests on the 1B loop at 2750 psig after replacing the coolant pump gasket and on the 1D steam drum and header at 1155 psig following repairs to steam valves.

May 2 — Using the turbine as a heat sink, 1A and 1C loops were cooled to 235°F to match 1B and 1D loop temperatures. 1B and 1D loops were then returned to service.

May 3 — Reactor taken critical to heat-up plant from 235 to 500°F with 1A, 1B, 1C, and 1D loops in service.

May 4 Performed the following tests:

a) Determination of reactor coolant system pressure drop.
b) Reactor protection system.
c) Flow distribution across the core.
d) Failed element detection and location system check-out.
e) 1D reactor coolant pump operational check.

May 5 Reactor critical at 9:02 A. M.; turbine brought to synchronous speed for annual overspeed trip insurance inspection. After several adjustments to the overspeed trip mechanism the turbine tripped at 1950 rpm.

May 6 Power operation for PWR Core I Seed 2 started when the main generator breaker was closed at 2:55 A. M. The station operated with all four reactor coolant loops in service.

At 10:51 P.M., the first safety shutdown occurred while the shift reactor engineer was adjusting nuclear instrumentation power amplifier gain settings to equate the reactor power level indicated by the nuclear instrumentation to the measured thermal output of the reactor. The test current signal on Channel D was inadvertently increased beyond 118% which was sufficient to energize the safety shutdown relays. A review of the adjustment procedure was discussed with all operating personnel to avoid a similar incident in the future.

May 7 Reactor taken critical; generator synchronized at 1:51 A. M.; station load increased to 100 percent reactor power for a 150-hour period of steady-state operation tests.

May 8-11 Station operated at 100 percent reactor power and the following tests were performed:

a) Reactivity lifetime.
b) Reactor pressure drop and coolant flow characteristics.
c) FEDAL system.
d) Periodic reactor plant leak rate.

May 12 Performed rapid station shutdown at 9:45 P. M. in preparation for 60-hour xenon transient test. 1D reactor coolant pump was shut down for testing; 1D loop stop valves remained open.

May 13-14 Performed xenon transient test.

May 15 Generator was synchronized at 5:13 P. M. and station load was varied for reactor coolant fission product activity test and reactor pressure drop coolant flow characteristics test. Reactor coolant loops 1A, 1B, and 1C were in service with 1D loop stop valves open and 1D pump shut down for testing.

May 16 — Station load was increased to 100 percent reactor power using only one boiler feed pump to carry full load for an operational test of the pump. After establishing that one boiler feed pump could carry full load without difficulty, the second boiler feed pump was placed in service in accordance with approved operating procedures.

May 17-21 — Station operated at full load for reactivity lifetime test.

May 22 — At 8:41 A.M., completed successful weekly test of turbine solenoid and simulated overspeed throttle trips. At 8:42 A.M., turbine throttles closed due to failure of the manual latch to engage at completion of testing. Reactor coolant system temperature and pressure were maintained within operating limits by inserting control rods and by operation of the decay heat relief valve. When the main generator breaker was manually opened, the automatic 2400 v bus transfer resulted in momentary loss of power to the 1A and 1C reactor coolant pumps causing a normal loop status safety shutdown.

Reactor taken critical; generator synchronized at 12:58 P.M.; station load increased to 100 percent reactor power.

May 23-31 — Station operated at 100 percent reactor power for reactivity lifetime test.

JUNE, 1960

June 1-9 — Station operated at 100 percent reactor power with 1A, 1B, and 1C reactor coolant loops in service. 1D loop stop valves were open to the reactor and the 1D reactor coolant pump was shut down for testing.

June 10 — Performed a rapid station shutdown at 6:15 P.M. in preparation for xenon transient test.

June 11-12 — Performed xenon transient test.

June 13 — Completed xenon transient test and reduced plant pressure to 1200 psig for radiation survey of reactor vessel head.

Reactor taken critical. As a prerequisite for load drop tests the main unit turbine overspeed trip setting was checked twice at 1990 rpm.

June 14 — Generator synchronized at 7:10 A.M.; station load varied to meet Duquesne Light Company System demand.

Load drop tests started at 7:25 P.M. For this test, reactor power levels of 25, 50, 75, and 95 percent were established. After a short stabilization period, the main generator breaker was tripped to reduce generator output to zero. Data was obtained on the resulting transient on reactor and turbine plant systems. All systems responded smoothly and within operating limitations.

After completing load drop tests at 9:20 P.M. the reactor was shut down for periodic control rod drive mechanism test.

June 15-19 — Station shut down for the following tests:

a) Flow distribution across the core.
b) Valve operating system performance.
c) Intercalibration of temperature sensing elements.
d) FEDAL system.
e) Primary self-actuated relief valve operation.

June 15 — At 10:44 A.M., 1D reactor coolant pump was returned to service.

June 19 — At 11:00 A.M., 1D reactor coolant pump was tripped by the overcurrent relay during pump start-up for the performance of a flow coastdown test. The pump was satisfactorily started several times and operated until 8:00 P.M. when the pump was again tripped by the overcurrent relay. 1D pump was removed from service pending further investigation of pump trips.

June 20 — Using the main turbine as a heat sink, the reactor plant temperature and pressure were reduced to 200°F and 500 psig in preparation for control rod drive mechanism No. 12 stator water jacket assembly replacement. (On May 4, 1960, during the performance of the control rod latching procedure, an electrical ground was detected on rod drive mechanism No. 12. Because of the ground, periodic resistance measurements were taken to ground. On June 20, 1960, because of very low resistance to ground from its stator windings a decision was made to replace the stator-water jacket assembly of rod drive mechanism No. 12.)

June 22 — At 8:22 P.M., 1D reactor coolant pump was started to obtain data on its characteristics. During the test 1D pump was tripped by a phase overcurrent relay on five successive attempts to restart the pump in accordance with the test procedure.

At 10:54 P.M., 1D loop was isolated to investigate cause of pump failure to start.

June 23 — Reactor critical at 3:00 A.M. to increase plant temperature from 200 to 500°F.

At 3:32 A.M., a safety shutdown occurred due to low coolant pressure which resulted from overcorrection of a high-pressure condition caused by a rapid plant heat-up. Reactor critical at 5:00 A.M. to continue plant heat-up.

At 5:55 P.M., another safety shutdown occurred when an instrument technician inadvertently removed a lead from 1C T_h receiver indicator sensing element during resistance readings of a spare transducer. Reactor critical at 7:40 P.M.; generator synchronized at 8:45 P.M. and load increased for testing.

June 24 — At 6:20 P.M., while increasing load to 100 percent power, a safety insertion signal was received, followed at 6:24 P.M. by a safety shutdown during the investigation of the insertion. Both insertion and shutdown were caused by possible drift and changes in the settings of reactor protection system mag-amps. For more information, refer to the discussion of mag-amps given in Part IV, Chapter 1 of this report.

June 25 — Reactor taken critical; generator synchronized at 5:58 P.M. and load increased to 100 percent power.

June 26-30 — Station load varied due to network current limitations resulting from construction work being performed on the Duquesne Light Company 138 KV (Z-28) transmission system.

JULY, 1960

July 1 — Station operated at 100 percent reactor power with 1A, 1B, and 1C reactor coolant loops in service. 1D loop remained isolated for investigation of 1D pump trips.

At 6:38 P.M., following the successful weekly performance of the weekly main turbine simulated overspeed trip test, the throttle valves tripped without apparent cause. Control rods were inserted and the decay heat relief valve was opened to compensate for the sudden load drop. A safety shutdown did not occur because all station electrical power was transferred from station service to the Duquesne Light System prior to performing the simulated overspeed trip test.

After four unsuccessful attempts the throttle valves were finally latched without corrective maintenance and the turbine brought to synchronous speed. Simulated overspeed trip test was again performed successfully but this time the latch mechanism held; generator synchronized at 7:14 P.M. and load increased to 100 percent reactor power.

July 2-8 — Station operated at full reactor power for testing.

July 7-8 — Load varied because generator power factor and voltage were approaching high limits due to a 138 KV transmission line being out of service for construction work.

July 9	Reduced load to 15 percent and performed successful weekly main turbine trip test; however, at 4:45 P.M. the throttle valves tripped when the latch mechanism failed to engage.
	Plant temperature was reduced to 200°F using the main turbine as a heat sink for maintenance work on steam valves.
July 10	Reactor taken critical; plant temperature increased to 500°F and station operated from 12:29 P.M. to 6:11 P.M. at 100 percent reactor power for Duquesne Light Company System demand.
July 11-16	Plant used to train the students in the Nuclear Power Station Training Program in reactor start-up and power operation.
July 17	Plant temperature reduced to 200°F to permit returning 1D reactor coolant loop to service.
July 18	Reactor critical to increase 1A, 1B, 1C, and 1D loop temperatures to 500°F. 1B coolant pump was then shut down with 1B loop stop valves open to permit 3-loop power operation as required for steam generator performance test. Generator synchronized at 9:20 A.M. and load increased for testing. Because of excessive tube leakage, 1A feedwater heater was removed from service for inspection and tube replacement.
July 19-31	Station load varied for steam generator performance test.
July 23	1B and 1C feedwater heaters isolated due to excessive tube leakage in 1B heater.
July 24	1B and 1C feedwater heaters returned to service after plugging 1B heater leaking tubes.
July 28	At 11:17 P.M., a safety shutdown occurred during adjustment of the nuclear instrumentation power amplifier gain settings.

AUGUST, 1960

Aug. 1-5	Station operated with 1A, 1B, and 1D loops in service with 1B loop stop valves open and 1B pump shut down. 1A feedwater heater remained isolated for tube replacement. Station load was varied for steam generator tests.
Aug. 3	Completed steam generator performance test.
Aug. 5	Station shut down at 12:33 A.M. in preparation for returning 1B coolant pump to service. 1B pump was placed in service and reactor taken critical; generator synchronized at 4:36 A.M. and reactor load increased to 100 percent.

Aug. 6-12	Station load varied for determination of temperature coefficient test; variation limited by construction work on 138 KV transmission line (Z-28). Construction work on the line was completed August 11.
Aug. 13	Performed rapid station shutdown at 8:24 A. M. in preparation for xenon transient test.
	At 10:15 A.M., 1B loop was removed from service to repair a leak in the valve operating system 3-way selector valve for the 1B loop main hydraulic valves.
Aug. 14	Performed xenon transient test.
Aug. 15	1B loop returned to service at 12:47 A. M.; reactor taken critical; generator synchronized at 7:02 A. M. and load increased to 100 percent reactor power.
Aug. 16-17	Station operated at full load for reactivity lifetime test.
Aug. 18	Performed rapid station shutdown at 4:02 P. M. in preparation for xenon transient test.
Aug. 19	Generator synchronized at 12:39 A. M. and station load increased to 100 percent reactor power.
Aug. 20-31	Station operated at full load for reactivity lifetime. FEDAL system, and reactor pressure drop and coolant flow characteristics tests.

SEPTEMBER, 1960

Sept. 1-12	Station operated at 100 percent reactor power with 1A, 1B, 1C, and 1D loops in service. 1A feedwater heater remained isolated for tube replacement.
Sept. 6	1B and 1C feedwater heaters isolated because of badly leaking chemical cleaning connection valve on 1B heater.
Sept. 7	1B and 1C feedwater heaters returned to service after replacing chemical cleaning connection valve on 1B heater.
Sept. 9	Reached design lifetime blanket at 8000 EFPH.
Sept. 12	Performed rapid station shutdown at 12:30 A. M. in preparation for xenon transient test. 1A loop was isolated at 1:15 A. M. and cooled to 200^{o}F to permit maintenance on steam valves.
Sept. 13-17	Station shut down for maintenance and testing.

Sept. 13	At 10:03 A.M., a safety shutdown occurred when an instrument man accidentally disconnected the lead of a T_h receiver indicator.
	Plant temperature reduced to 200°F to permit maintenance on steam valves.
Sept. 14	1A boiler was drained and steam drum opened for vender's inspection.
Sept. 15	1A loop returned to service and boiler successfully leak tested.
Sept. 16	Reactor critical for plant heat-up to 500°F.
Sept. 17	With the reactor critical and 1A, 1B, 1C, and 1D loops in service at 500°F, a controlled safety test was performed. The purpose of the test was to obtain data on the dynamic response of the reactor plant to various control rod withdrawal transients. The plant and all safety devices operated properly.
Sept. 18-30	Station load varied for testing.
Sept. 18	At 3:48 A.M., with all control rods at a height of one inch for connection and calibration of special test instrumentation, a safety shutdown occurred. A technician working on the 1C loop T_h receiver erroneously disconnected the wires for the 1B loop T_h receiver; hence, when the 1C loop T_h instrument was calibrated, a scram ensued. The reactor was taken critical and the generator synchronized at 11:49 A.M. for another controlled safety test.
Sept. 22	Station shutdown at 8:03 P.M. for calibration of temperature sensing elements test.
Sept. 23-24	Performed reactor protection system time response test to measure delay time from loss of coolant pump power to opening of safety shutdown breaker.
Sept. 25	Reactor taken critical, generator synchronized at 3:31 P.M., and load varied for testing.
	At 3:45 P.M., a safety shutdown, due to a faulty microswitch on the T_{avg} receiver indicator, occurred when reactor power reached 50 percent. The microswitch, part of rod drop circuitry, was set for 490°F; however, it was faulty and was making contact with the average temperature at 500°F. Thus the shutdown occurred when reactor power reached 50 percent power. The switch was repaired, reactor taken critical, generator synchronized at 6:53 P.M., and load varied for testing.
Sept. 27	At 4:32P.M. with the station operating at approximately 100 percent power, the 1A loop was removed from service and cooled down in preparation for installation of special boiler instrumentation.

At this time the steaming level of 1D boiler during three-loop operation was established at +3.5 inches to minimize circulation difficulties believed to be existing in this type boiler.

OCTOBER, 1960

Oct. 1-13	Station operated at 100 percent reactor power with 1B, 1C, and 1D loops in service. 1A loop was isolated for calibration of d/p cells and steam drum modifications for steam generator performance tests. 1A feedwater heater remained isolated for tube replacement.
Oct. 13	Station was shutdown at 12:34 A.M. to return 1A loop to service. After returning 1A loop to service at 1:56 A.M., the 1B coolant pump was shut down. Steaming levels of 1A and 1D boilers were maintained at +3.5 inches for steam generator performance test.
Oct. 14-15	Station load varied for steam generator performance test.
Oct. 16-24	Station load maintained at 100 percent power for testing.
Oct. 17	1B loop isolated and cooled down for calibration of d/p cells.
Oct. 24	Station shut down at 12:16 A.M. and 1B loop returned to service.
	1B and 1C feedwater heaters isolated because of excessive tube leakage in 1B heater.
	Generator synchronized at 9:39 P.M. with 1A, 1B, 1C, and 1D loops in service; load varied for testing.
Oct. 25	With the station operating at approximately 20 percent power, 1B main steam stop valve was closed at 7:22 P.M. and 1C main steam stop valve closed at 11:23 P.M. for core power distribution tests.
Oct. 26	Station shut down at 7:56 P.M. in preparation for training. 1A loop was isolated and cooled down for additional steam drum modifications and d/p cell repairs.
Oct. 27	1B and 1C feedwater heaters returned to service after plugging leaking tubes in 1B heater.
Oct. 28-31	Plant used to train the students in the Nuclear Power Station Training Program in reactor start-up and power operation.

NOVEMBER, 1960

Nov. 1-2	Station operated with 1B, 1C, and 1D loops in service for reactor start-up and power operation training required of students in the Nuclear Power Station Training Program. 1A loop was isolated for additional steam drum modifications and d/p cell repairs. 1A feedwater heater remained isolated for tube replacement.
Nov. 2	Generator synchronized at 12:57 P.M. and load increased for testing.
Nov. 3-25	Station operated at 100 percent power for reactivity lifetime test.
Nov. 6	Station shut down at 12:45 A.M. to return 1A loop to service. Generator synchronized at 6:15 A.M. and load increased for testing.
Nov. 8	1D coolant pump was shut down at 1:48 A.M. for testing.
Nov. 10	1D loop was isolated at 4:00 P.M. and cooled down for steam drum modifications and calibration of d/p cells.
Nov. 23	Performed successful leak test on 1D steam drum at 1150 psig.
Nov. 25	Performed rapid station shut down in preparation for xenon transient test.
Nov. 26-30	Station shut down for maintenance and testing.
Nov. 26	1B and 1C feedwater heaters again isolated due to excessive tube leakage in 1B heater.
Nov. 28	Performed control rod positions for criticality test.
Nov. 29	Plant temperature reduced to 135°F for maintenance and inspection of all steam drums and flash, blow-off, and gravity drain tanks.
	1B and 1C feedwater heaters returned to service after plugging leaking tubes in 1B heater.
Nov. 30	1B boiler feed pump (rated at 2300 volts) was satisfactorily operated with power supplied from the 23 KV emergency power transformer as a test of this method of operating the pump for safety injection purposes during a loss of normal station AC power.

DECEMBER, 1960

Dec. 1-5	Station remained shut down for maintenance and testing. 1A feedwater heater remained isolated for tube replacement.
Dec. 1	Plant at 170°F and 400 psig. 1D loop returned to service following completion of steam drum modifications and calibration of d/p cells.
Dec. 2	Leak tested all steam drums satisfactorily at 1150 psig. Performed control rod positions for criticality test and reactor plant container air cooling system 48-inch butterfly valve leak test.
Dec. 3	Plant pressure varied between 250 psig and 2000 psig for periodic calibration of pressure instrumentation test. The reactor plant cooldown and temperature control system was placed in service to maintain temperature at 200°F.
Dec. 4	Started plant heat-up with reactor critical for temperature coefficient of reactivity tests.
Dec. 5	Plant temperature 500°F; temperature coefficients of reactivity test completed.
Dec. 6	Reactor taken critical; generator synchronized at 2:12 A.M. with all four loops in service; load increased for testing.
Dec. 7	Because of suspected leakage in 1A loop heat exchanger the 1A loop was isolated and cooled down in preparation for heat exchanger tube sheet inspection and leak testing. At 11:47 A.M., safety shutdown occurred when a faulty amplifier on 1D loop hydraulic valve position indicator initiated a "valve drift" signal causing a loop status safety shutdown. After repairing the valve position indicator amplifier the reactor was taken critical, generator synchronized at 5:26 P.M., and load increased for testing.
Dec. 8-31	Station operated at 100 percent reactor power with 1B, 1C, and 1D loops in service for power operation and reactivity lifetime test.
Dec. 25-26	Station load reduced to 30 MW (net) to meet requirements of Duquesne Light System load schedule.
Dec. 31	At 9:42 A.M., a safety insertion occurred during operation of the reactor protection system test set when the test selector switch was accidently moved from the "test" to "operate" position while Channel D of the nuclear instrumentation system was set at 114 percent by the test signal circuit.

Generator load was reduced to approximately 80 percent reactor power to maintain average coolant temperature at 500°F. After conditions stabilized, station was returned to 100 percent reactor power to continue testing.

JANUARY, 1961

Jan. 1-27	Station operated at 100 percent reactor power with 1B, 1C, and 1D loops in service for power operation reactivity lifetime test. 1A loop was isolated to investigate primary-to-secondary leakage in the 1A heat exchanger.
Jan. 6	Load reduced to 90 percent reactor power during cleaning of main unit turbine condenser tubes.
Jan. 22	Load reduced to isolate 1B and 1C feedwater heater due to excessive tube leakage. Following the removal of the heaters, load was increased to 100 percent reactor power.
Jan. 24	1B and 1C feedwater heaters returned to service after plugging leaking tubes in 1B heater and load increased to 100 percent reactor power.
Jan. 28	Performed rapid station shutdown at 5:06 P.M. as per test procedure; however, during the shutdown, average coolant temperature continued to increase although the decay heat relief valve had been opened and control rods were being manually inserted. A safety shutdown occurred when T_h reached 522°F. The cause of the scram was due to a combination of low rod worth in the area of the controlling group and the diminished effect of the negative temperature coefficient.
Jan. 29-31	Station shut down for maintenance and testing.
Jan. 29	Performed xenon transient test.
Jan. 30	Reduced plant pressure to 1200 psig for a radiation survey of the reactor vessel head. At 5:22 P.M., a fire was reported on the reactor vessel head. The cause of this fire was smoldering anti-contamination clothing which had been used by radiation survey personnel for insulation on two refueling ports. During the fire No. 53 rod drive mechanism temperature increased to 200°F, but decreased to normal (90°F) after the fire was extinguished at 5:45 P.M. An extensive investigation for the extent of damage to power and control cables in the vicinity of the fire was conducted.
Jan. 31	Replaced thermocouple on rod drive mechanism No. 53 because a resistance check of the old thermocouple showed erratic readings. Performed periodic control rod drive mechanism test.

FEBRUARY, 1961

Feb. 1 — Station remained shut down for maintenance and testing. 1B, 1C, and 1D loops were in service with 1A loop isolated for heat exchanger leak test.

1A feedwater heater was made available for service after replacing all tubes. Main turbine trip mechanism was inspected to determine the cause of unreliability of the relatch mechanism under load. The 1B loop was isolated and cooled down for steam valve maintenance.

Feb. 2 — Completed periodic control rod drive mechanism test.

Feb. 3 — 1B loop returned to service, reactor taken critical, and main turbine overspeed trip checked at 1940 RPM for annual insurance inspection. Generator synchronized at 11:12 P.M. and load increased for testing.

Feb. 4-18 — Station operated at 100 percent reactor power with 1B, 1C, and 1D loops and 1A, 1B, and 1C feedwater heaters in service for reactivity lifetime test.

Feb. 17 — Station load reduced to 90 percent reactor power to isolate 1B and 1C feedwater heaters because of leakage around 1C heater head gasket.

The head gasket was replaced, both heaters returned to service, and load increased to 100 percent power.

Feb. 19 — Station shut down at 12:38 A.M. to return 1A loop to service after heat exchanger repairs and leak testing were completed. 1D coolant pump thermocouple was also replaced because of improper indication. Generator synchronized at 2:47 P.M. and load increased for testing.

Feb. 20-28 — Station operated at 100 percent power for power production and reactivity lifetime test except during the time it was necessary to reduce load to isolate 1B and 1C feedwater heaters.

Feb. 21 — Load was reduced to 90 percent to again isolate 1B and 1C feedwater heaters to repair leaking 1C heater head gasket.

Feb. 23 — Leakage in 1B feedwater heater was 17.5 gpm as determined during the leak test of the heaters following 1C heater repairs. 1B and 1C feedwater heaters were returned to service and load increased to 100 percent power.

MARCH, 1961

Mar. 1-29 — Station operated at 100 percent power for power production and reactivity lifetime test. All four reactor coolant loops including 1AC and 1BD purification loops were in service. Because of coolant leakage through air operated charging valves, first detected by a pressure build-up equal to plant operating pressure on the charging pump discharge piping, reactor plant charging was limited through one purification loop during the month until repairs could be made to both charging valves, one at a time. By the end of the month both 1AC and 1BD charging system air operated globe valves were repaired; special leak tests performed on the valves indicated that the repairs were satisfactory.

Mar. 30 — Performed rapid station shutdown at 4:03 P.M. in preparation for xenon transient test. At 7:32 P.M., all control rods were fully withdrawn, unable to override xenon.

Mar. 31 — At 10:00 A.M. with all control rods fully withdrawn the reactor was again critical for xenon transient test.

APRIL, 1961

Apr. 1 — Station remained shut down for testing. 1A, 1B, and 1D loops were in service with 1C loop stop valves and 1C pump shut down. 1B and 1C feedwater heaters were isolated due to 45 gpm leakage in 1B heater. The xenon transient test was completed and 1C coolant pump returned to service. Performed radiation survey of reactor vessel head and calibration of temperature sensing elements tests.

Apr. 2 — A test of the reactor plant cooldown and temperature control system was performed; however, the system did not operate as designed.

Started plant cooldown using main turbine as heat sink in preparation for maintenance and operational training.

Apr. 3 — Plant at 180°F and 400 psig. Performed radiation survey 100 hours after station shutdown.

Apr. 4-5 — With the plant at 180°F and 400 psig and the reactor shut down, the following emergency drills were performed by each shift:

a) Loss of normal and all AC power.
b) Loss of all AC power.
c) Simulated safety injection.
d) Loss of component cooling water.
e) Malfunction of control rods.

Apr. 6 — The condensate storage was removed from service after it was discovered that approximately one-third of the dome top of the condensate storage tank collapsed due to failure of the vacuum breaker to operate correctly when the storage tank was subjected to a negative pressure.

Reactor taken critical for plant heat-up, control rod positions for criticality test, and temperature coefficients of reactivity test. 1B and 1C feedwater heaters were made available for service.

At 9:46 P.M., a safety shutdown occurred at a flux level of 5×10^{-7} amps (recorder scram set point for low-power testing) caused by misalignment of Channel B nuclear instrumentation system recorder.

Apr. 7 — Reactor taken critical and plant temperature increased to 500°F.

Apr. 8 — Another test of the reactor plant cooldown and temperature control system was performed; however, the system did not operate properly. It was decided not to operate the system above 250°F until further evaluation was conducted.

Performed initial starting current and operating tests on 1D coolant pump.

Apr. 9 — 1D coolant pump returned to service.

Apr. 10-13 — Plant used to train students in the Nuclear Power Station Training Program in reactor start-up and power operation.

Apr. 14 — Reactor taken critical; generator synchronized at 1:51 A.M. with all four loops in service; load increased to 100 percent power for testing.

Apr. 15-30 — Station operated at 100 percent power for power production and reactivity lifetime test.

Apr. 19 — At 11:13 P.M., a safety insertion occurred due to a bistable mag-amp drift. The mag-amp was readjusted to the normal firing point. Refer to Part IV, Chapter 1 of this report for further details of mag-amp drift problems.

Apr. 20 — At 4:20 A.M., another safety insertion occurred due to mag-amp drift. As a result of the safety insertions, reactor power was reduced 3 percent to permit mag-amp drift tests.

Started calibration of individual feedwater flow integrators to verify the validity of calorimetrics based on total feedwater flow.

Apr. 26 — Because of the individual feedwater flow integrators were being calibrated, and since the total feedwater flow integrator is physically located on the inlet to the 1B and 1C feedwater heaters which were known to be leaking thus indicating a greater total feedwater flow, it was decided to use primary calorimetrics to calibrate the nuclear instrumentation system.

Apr. 27 — Load reduced to isolated 1B and 1C feedwater heaters due to excessive tube leakage. No leaks in 1C heater; however, 1B heater leakage was 145 gpm.

Apr. 28-30	Station operated at 100 percent reactor power (54 MW net) due to 1B and 1C feedwater heaters being isolated.

MAY, 1961

May 1-31	Station operated at 100 percent reactor power with 1A, 1B, 1C, and 1D loops in service for power production and reactivity lifetime test. 1B and 1C feedwater heaters were isolated because of 1B heater tube leakage.
May 1	Completed boiler feedwater integrator calibration and again started to use secondary calorimetrics to adjust nuclear instrumentation.
May 3	1B and 1C feedwater heaters returned to service after plugging leaking tubes in 1B heater and load increased to 100 percent reactor power.
May 4	Total feedwater integrator calibrated.
May 12	Isolated the component cooling water cooler because of dirty tubes and placed the canal water cooler in service on the component cooling water system. The component cooling water and canal water coolers are interconnected so that either cooler can be used in either system.
May 17	At 1:12 P.M., a safety insertion occurred while an instrument technician was investigating "spikes" on a nuclear instrumentation recorder. An investigation revealed that the insertion was caused by interaction between linear power amplifiers due to aging of the auctioneering diodes. This condition was corrected by replacing all the diodes in the nuclear instrumentation system auctioneering circuitry. The reactor was returned to full power.
May 22	Because of a bad feedwater leak at the 1A boiler feedwater regulating valve flange, 1A reactor coolant pump was shut down at 3:09 P.M. and the 1A main steam M.O. stop valve was closed. 1A loop stop valves remained open.
May 23	Reduced load to 90 percent reactor power at 8:48 P.M. for axial flux perturbation test. The purpose of the test was to measure the reactivity fluctuation resulting from a non-equilibrium axial xenon distribution induced by altering the control rod configuration while at 90 percent power.
May 25	Condensate storage tank made available for service following completion of repairs. At 11:03 P.M., 138 KV transmission line Z28 oil circuit breaker tripped due to pilot wire relay operation. Z28 oil circuit breaker was again closed at 11:04 P.M. at Duquesne Light System Operator's request. The loss of the 138 KV transmission had no effect on the reactor.
May 26	Completed repairs on 1A boiler feedwater regulating valve.

May 27 — Immediately after completing the axial flux perturbation test the control rods were returned to their normal programming sequence and station load was increased to 100 percent reactor power.

Discovered main unit air ejector tube leakage as indicated by increased drainage (approx. 300 gal/hr) from air ejector after condenser drain line. The air ejector tube leakage had no effect on condenser vacuum.

JUNE, 1961

June 1-10 — Station operated at 100 percent reactor power with 1B, 1C, and 1D loops in service for power production and reactivity lifetime test. 1A reactor coolant pump was shut down with 1A loop stop valves open. Main unit air ejector tube leakage remained at approximately 300 gal/hr.

June 5 — Because of an increase in main unit air ejector tube leakage to approximately 850 gal/hr the air ejector drain line was cut to allow for the added drainage and thus prevent a reduction in condenser vacuum.

June 7 — At 1:13 P.M., all control rods were withdrawn to the upper programming limit (69 inches). To obtain an additional 100 EFPH from Core I Seed 2 rod sub-groups 1 through 7 were withdrawn to the full travel limit and sub-group 8 was inserted to 64.75 inches to maintain 500°F average coolant temperature.

June 10 — The end of full power operation of PWR Core I Seed 2, 7528.3 equivalent full power hours, was reached at 11:02 A.M. after allowing plant temperature to decrease to 497°F with all control rods completely withdrawn. The station was shut down. This station shutdown also ended a continuous full power run of 57 days for 1357 EFPH.

The reactor was shut down and 1A reactor coolant pump was returned to service at 1:52 P.M.

June 11-19 — Station shut down for maintenance and testing.

June 11 — At 10:05 P.M. with all control rods withdrawn to their full travel limit the reactor was again critical for xenon transient test.

June 12 — Shut down reactor after completing xenon transient test and at 4:05 A.M. started plant cooldown by venting steam from main steam leads in preparation for steam valve maintenance and cold plant testing.

June 13 — With plant at 250°F placed reactor plant cooldown and temperature control system in service to reduce plant temperature to approximately 120°F.

At 9:54 A.M. with the plant at 450 psig and 148°F the Shift Reactor Engineer, while checking the pressurizer pressure indicator, accidently moved the dial indicator to the pilot operated relief valve setting causing the relief valve to open and a drop in plant pressure to 100 psig. All reactor coolant pumps were immediately shut down. All pumps were meggered, vented, and found satisfactory. 1A, 1B, and 1C pumps were returned to service to continue plant cooldown.

At 6:37 P.M. 1A reactor coolant pump was shut down to maintain plant at 130°F.

Four tubes in the main unit air ejector were found to be leaking. All four tubes were replaced.

June 14 — Performed control rod positions for criticality test. 1D loop was isolated at 8:50 A.M. in preparation for coolant pump and volute removal.

June 15-17 — Performed calibration and intercomparison of control rods test.

June 15 — At 9:40 A.M., while performing calibration and intercomparison of control rods test, a safety insertion occurred as a result of a high start-up rate signal on Channel D nuclear instrumentation system intermediate range. The trouble was traced to a defective compensated ion chamber cable connection.

June 16 — 1D coolant pump was removed from the boiler chamber to the canal area.

June 18 — Started plant heat-up to 235°F with 1A, 1B, and 1C coolant pumps for coefficients of reactivity test.

June 19 — Plant solid at 235°F and 500 psig. Established pressurizer bubble and brought reactor critical to heat up plant to 450°F. From 450°F the reactor coolant pumps were used to increase plant temperature for coefficients of reactivity test.

June 20 — Changed reactor instrumentation and reactor protection system set points to agree with a new plant operating temperature of 475°F as required for reduced power operation at the end of Seed 2 life.

1B and 1C feedwater heaters were isolated to repair 1B heater tube leakage.

June 20-27 — The reactor was started up, the main generator synchronized, and loaded to about 25 percent power 58 times for FEDAL system transient testing during plant start-up.

June 21 — At 8:55 P.M., a safety shutdown occurred at 22 percent power during a station start-up. The shutdown was caused by the source range selector switch being positioned between the "in" and "pull-out" position. Investigation revealed that with this condition cold plant protection (24 percent) remained in service.

June 22 — At 3:34 A.M., another safety shutdown occurred at 22 percent power while increasing load rapidly for testing. The operator did not have sufficient time to pull out the source range selector switch to remove cold plant protection before reaching 24 percent power. The test procedure was changed to allow for this condition.

June 28 — Performed station on-site emergency drill for minor accidental release of radioactivity to the atmosphere. The drill was satisfactory.

June 29-30 — Plant temperature was varied between 400 and 530°F for periodic calibration of temperature sensing elements tests.

<u>JULY, 1961</u>

July 1-3 — Station was shut down for testing. 1A, 1B, and 1C loops were in service. 1D loop remained out of service for pump volute removal. 1B and 1C feedwater heaters were out of service to repair 1B heater tube leaks. Plant at 475°F and 1800 psig.

The following tests were performed:

a) Control rod drive mechanism periodic test.
b) Steam generator blowdown rate.
c) Radiation survey of reactor vessel head.

July 4 — Reactor brought critical; main generator synchronized at 9:25 P.M. and load increased to 75 percent reactor power for an extended end of seed life power run.

July 5-6 — Load maintained at 75 percent power for testing.

July 7-15 — Performed a radial flux tilt test to determine the variation of certain plant parameters when a radial power tilt is introduced across the core by deliberately misaligning diametrically opposed control rods.

July 16 — Performed rapid station shutdown at 5:32 A.M. after all control rods were completely withdrawn and plant temperature decreased to 472°F.

July 17-23 — Station shut down for 250-hour samarium transient test.

July 17 — Reactor critical at 12:23 P.M.

July 19 — At 8:20 A.M., a low pressure safety shutdown occurred when an instrumentation technician, troubleshooting the pressurizer recorder differential transformer by manually moving the core of the transformer, transmitted a signal back to the wide range pressurizer instrument causing its indicator to deflect downward past the low pressure scram contact setting.

July 22 — At 3:12 A.M., a safety shutdown occurred when the recorder scram set point of 5×10^{-7} amps for non-power testing was exceeded due to a spurious full scale deflection of channel B nuclear instrumentation system recorder. No other nuclear instrumentation recorder showed a similar spike. The recorder was cleaned and repaired and returned to service.

Reactor taken critical to continue testing.

July 23 — Xenon transient test completed and reactor shut down with all rods on the bottom at 6:25 A.M.

July 24-29 — Performed reactor start-ups and power operation on each shift for Duquesne Light Company operational training.

July 27 — At 4:13 P.M., a safety insertion occurred while increasing station load for training purposes. Inspection of the reactor protection system indicated a condition of high power for existing flow (less than three pumps in fast speed). However, since the three reactor coolant pumps were actually in fast speed as indicated by their pump speed selector switches, loop flows, pump currents, and pressure drops the signal to the reactor protection system was false. The trouble was traced to poor contact on the speed changer auxiliary switch.

July 30 — Reactor taken critical; main generator synchronized at 12:02 P.M. and load increased to approximately 50 percent reactor power and held constant for another extended end of seed life power run and testing.

July 31 — Maintained load at 50 percent power.

AUGUST, 1961

Aug. 1-12 — The final extended power run of Core I Seed 2, which began July 30, ended August 12, 1961 at 12:06 A.M. with 7900.6 equivalent full power hours of operations. The station was shut down and preparations started for the refueling operation.

Aug. 12-13 — Maintained plant at 500°F to perform primary plant self-actuated relief valve operation test and periodic reactor plant leak rate test.

Aug. 14	Reactor taken critical; main generator synchronized at 11:36 A.M. and loaded to 30 percent reactor power for Duquesne Light Company operational training. The station was shut down at 12:05 P.M. with a final total of 7900.7 EFPH of operations on Seed 2 and 13,707.0 EFPH on the blanket. Periodic reactor plant leak rate test and control rod drive mechanism exercise tests were performed.
Aug. 15	Started plant cooldown using the turbine as a heat sink. Performed flow distribution across the core tests at plant temperatures of 200 and 300°F.
Aug. 16	Placed 1AC and 1BD core removal cooling systems in service to reduce plant temperature to 100°F; posted refueling clearances; layed-up turbine plant systems under a nitrogen blanket and started reactor refueling.

CHAPTER 4

OPERATING INCIDENTS

Introduction

Prior to the operation of the Shippingport Atomic Power Station, a system was instituted for reporting the details and corrective action taken to cope with emergency or unusual conditions. The primary purpose of this reporting system was to disseminate information on each incident as rapidly as possible to persons associated with the Shippingport Atomic Power Station in order to prevent the occurrence of similar incidents. It is to be noted that this reporting system places major emphasis on direct reactor plant incidents or Turbine-Generator incidents which effect the entire plant.

Incidents in the following categories are reported:

1. Reactor Plant Primary and Auxiliary Systems and Equipment.

   a. Subjection of equipment or systems to conditions which exceed specified operating or design limitations.

   b. Safety shutdowns or safety insertions excluding those deliberately initiated for training or testing.

   c. Operational errors which could damage equipment or systems.

   d. Significant malfunctions of components, piping systems, or electrical systems.

   e. Incidents considered to be of interest to other facilities.

2. Turbine-Generator Plant Systems and Equipment.

   a. Direct effect on over-all plant control.

   b. Direct effect on reactor plant control or operation.

   c. Damage to or effect on the integrity of reactor plant systems or equipment.

Each incident report is, therefore, assigned a general designation of apparent cause to one, or a combination, of the following categories: Design, material, personnel, or procedure. Of the 43 incidents which occurred during Core I Seed 2 operation, the general designation of apparent cause was as follows:

Single Cause

Design	2	Material	15	Personnel	12
Procedure	2	Unknown	2		

More Than One Cause

Design and Material	3	Design and Procedure	1
Personnel and Procedure	1	Material and Personnel	2
Procedure, Material, and Personnel	1	Design, Personnel, and Procedure	2

Thirty-six of the 43 incidents involved systems and equipment associated with the reactor plant. Six incidents were associated with the systems and equipment of the turbine generator portion of the station. The remaining incident was the result of a fire on the reactor vessel head. Fourteen incidents occurred while the station was at power and resulted in either complete loss or reduction of station capacity. Of these 14 incidents, seven resulted from safety shutdowns, five were safety insertions, and two were the result of main unit turbine trips.

Discussion

A few of those incidents which were unusual or uniquely informative are described below in summary form. A complete tabulation of all incidents with a brief description and the designated cause is given in the following pages.

Electrical Ground on Control Rod Mechanism

On May 2, 1960, during the performance of the reactor control and instrumentation precritical check, an electrical ground was detected on the 230-volt rod drive power supply when the rod drive mechanisms were engaged. By isolating segments of the rod control system, it was determined that the fault existed in the circuitry associated with control rod no. 12. The reactor rod control system was de-energized and the resistance between the rod no. 12 power cables and ground was found to be 2400 ohms. By measurement directly into the Amphenol connector on the starter housing, it was ascertained that the ground existed within the housing. Rod no. 12 was then energized by 140 volts from the alternate power supply for two hours in an attempt to "dry out" the ground. After this period, the resistance to ground increased to 35,000 ohms. A minimum value of one megohm is considered satisfactory although lower values can be tolerated.

Subsequent to reactor shutdown on June 13, 1960, resistance measurements to ground taken on rod drive mechanism no. 12 showed that the power winding insulation strength had deteriorated with respect to the length of time de-energized.

Because of this apparent ground, it was decided to replace rod drive mechanism stator no. 12. The defective stator was replaced with a spare on June 21, 1960. The defective stator assembly was returned to Bettis where both hydrostatic and helium leak tests were performed on the water jacket; no leaks were discovered. The stator was then dried but still showed a solid ground of less than 0.1 ohm resistance, indicating that the problem was not one of moisture alone.

The failure of the mechanism stator winding could have been caused by a spray of cooling water in the area of this mechanism while the watertight caps were not on the Amphenol connectors during refueling. The wet condition may have led to the deterioration and failure of the insulation.

Since a spray incident similar to the one causing this failure could not occur during normal operation, no design changes were made.

Reactor Coolant Pump Gasket Leakage

On two occasions leakage was detected around the periphery of the gasketed flange between the reactor coolant pump stator assembly and the volute. In both cases, the loop and pump were being hydrostatically tested at 2750 psig.

In April, 1960, following refueling, leakage occurred in the 1B reactor coolant pump. Subsequently, the lengths of four holddown bolts (approximately 90 degrees apart) were measured while the bolts were installed, and again after they were removed and cooled to determine whether the bolts were properly torqued. These measurements disclosed that the existing elongation was 5 to 7 mils under the prescribed amount. The bolts were then reinstalled and torqued enough to stretch them to a length 5 to 7 mils greater than the "as found" length. The remaining 20 bolts were also heated and stretched an additional 5 to 7 mils to return all bolts to their proper settings.

A re-evaluation of the maximum stress that could be exerted on the pump flange at this stage dictated that the bolt torques be decreased. Since it was possible that the gasket would then leak, all bolts were removed, a new gasket installed, and the bolts retorqued. After a successful hydrostatic test at 2750 psig, the 1B loop was returned to service.

The copper gasket removed from the 1B reactor coolant pump was returned to the Bettis Atomic Power Laboratory for destructive testing and evaluation. The evaluation disclosed that no discernible creep was present in the material. However, a slight degree of workhardening due to compression bolt loading was evident. The evaluation also indicated that the gasket could have lasted for an indefinite number of bolt tightenings had it been left in place.

The pump manufacturer stated that the bolt relaxation was caused by a readjustment of stresses with associated yielding in the bolt and flange threads of the pump. Furthermore, the copper gasket material did not contribute to the bolt relaxation.

A revised procedure based upon these evaluations was subsequently developed.

1. The flange bolts should not be torqued unless an actual leak occurs.

2. If a leak occurs during a hydrostatic test, the test should be terminated and the bolts restretched to 0.014 $\pm$ 0.001 inch (provided they are less than this amount) and the hydrostatic test repeated.

3. If restretching the bolts does not stop the flange leak, the gasket should be replaced.

The procedure of bolt tightening stated in step 2 can be repeated an indefinite number of times insofar as the copper gasket material is concerned since the gasket material retains a relatively soft core, the degree of workhardening on the gasket being very slight.

In January of 1961 when the 1A reactor coolant pump gasket leakage was observed, the pump holddown bolts were removed, heated, reinstalled, and stretched in groups of four to obtain a satisfactory flange seal.

Turbine Throttle Trips

In May, 1960, following the successful weekly test of the main turbine solenoid and simulated overspeed trip, the auto-stop-reset lever was returned to the normal operating position. When the reset lever was released, the overspeed trip latch failed to engage the overspeed trip valve stem sleeve resulting in the tripping of the main turbine throttle valves, thus shutting down the turbine.

Again on July 1, 1960, a turbine trip occurred after the successful weekly test of the simulated overspeed trips and main turbine solenoid. A study of the problem resulted in a thorough check of the throttle trip mechanism. Measurements were taken which indicated that the clearance between the overspeed trip body and the trip trigger, with the turbine throttle trip mechanism latched, was less than the required minimum of 0.062 inches. The trip trigger was removed and the face of the tripper was ground to increase the clearance between the trip trigger and the overspeed trip body to 0.085 inches. The mechanism was reassembled, a simulated overspeed trip test was performed, and the latch successfully reset at 1800 rpm with no generator load. The same test was performed with the station operating at 60 MW net electrical output and the latch was again successfully reset. At this time, the turbine solenoid trip, initiated from the turbine console, was also actuated and the trip mechanism was successfully relatched.

Weekly tests of the main turbine simulated overspeed trip, which were postponed due to the turbine throttle trip difficulties, were resumed. Subsequent operation proved the repairs to be satisfactory.

Duquesne Light Company
Shippingport Atomic Power Station

Incidents

April 2, 1960 - August 15, 1961

Date	Basic Cause	Plant Operating Conditions	Remarks
April 2, 1960	Design Procedure	Station shut down - Reactor Protection System test	A time delay in excess of the specified limits was revealed by measuring the time response between introduction of the initiating signal and the safety shutdown relay operation. The apparent excessive time delay was caused by the test method being improperly specified.
April 4, 1960	Procedure	Station shut down - physics test	A safety shutdown was initiated through NIS Channel C recorder contact. A discrepancy of one decade existed between the NIS recorder and the console meter.
April 15, 1960	Material	Station shut down - physics test	During the performance of a physics test, control rod 12 was dropped from 69 in. while transferring it from inverter no. 9 to inverter no. 1.
April 20, 1960	Material	Station shut down - physics test	Operational check of the no. 2 spare inverter; hold resistance switch (S-106) was smoking, the contacts severely charred and partially melted.
April 22, 1960	Design	Station shut down - maintenance	Hydrostatic testing on the 1B loop at 2750 psi revealed leakage at the gasketed joint of the 1B reactor coolant pump.

Date	Basic Cause	Plant Operating Conditions	Remarks
April 27, 1960	Material	Station shut down - physics test	Steam leak in one of the 1D main steam lead fillet welds at operating temperature and pressure.
May 2, 1960	Unknown	Station start-up	Electrical ground was detected on the 230-volt rod drive power supply when the rod drive mechanisms were engaged.
May 4, 1960, June 13, 1960	Material	Station start-up	Electrical ground was detected on the 230-volt rod drive power supply when rod drive mechanism no. 12 was energized.
May 5, 1960	Material	Station start-up	Galled valve steam threads on FEDAL System sample interconnection valve.
May 6, 1960	Personnel	Station operating - 60 MW	Safety shutdown resulted when the NIS Channel D linear amplifier test signal was increased beyond 118% during NIS gain setting adjustment.
May 7, 1960	Material	Station operating - 60 MW	Two safety insertions resulted from rod withdrawal at 103% power while attempting to maintain operating reactor coolant temperature.
May 22, 1960	Material	Station operating - 60 MW	Main turbine throttle valves tripped when the overspeed latch failed to engage following simulated overspeed.
May 27, 1960	Design Material	Station operating - 60 MW	The river water booster pump discharge strainer inlet valve became jammed when the valve seating gasket dislodged, thus preventing strainer isolation.

Date	Basic Cause	Plant Operating Conditions	Remarks
June 19, 1960	Personnel Material	Station shut down - flow test	1D reactor coolant pump A. C. B. tripped open seconds after closure. Several pump A. C. B. trips had been experienced due to faults in the special pump tripping circuits installed for the flow coastdown test. The cause of the trip was later revealed to be the A and C phase overcurrent relays.
June 23, 1960	Personnel	Station start-up	Low coolant pressure actuated safety shutdown subsequent to reducing from high-pressure condition by operating pressurizer spray and inserting control rods.
June 23, 1960	Personnel	Station shut down - maintenance	High T_h safety shutdown initiated subsequent to the exchange of spare and normal 1C loop T_h thermometer signal leads between the Norwood and Bristol Recorders, when the spare T_h leads were also disconnected.
June 24, 1960	Material Personnel	Station operating - 60 MW	A 105% power level safety insertion occurred. While attempting to ascertain the cause of the safety insertion, a maximum neutron flux level safety shutdown resulted at 112%. These incidents were partially the result of bistable magnetic amplifier drift.
July 1, 1960	Material	Station operating - 60 MW	Main turbine throttle valves tripped following solenoid and overspeed trips prior to releasing reset lever without apparent cause.

Date	Basic Cause	Plant Operating Conditions	Remarks
July 28, 1960	Personnel	Station start-up	A +6 in. discrepancy on rods 62 and 63 from the indicated Group II bank height was noted on the rod position dial indicators. This height difference was attributed to failure to rezero dial indicators during latching operations.
July 28, 1960	Personnel	Station operating - 23 MW	NIS coincidence safety shutdown occurred during NIS drift check when the test signal for Channel D was not reduced below the drop out point and Channel C was subsequently drift checked initiating the shutdown.
Sept. 13, 1960	Personnel	Station shut down - physics test	Safety shutdown resulted when the 1C loop T_h resistance thermometer leads were accidently removed during maintenance work, driving the Norwood receiver-indicator up-scale.
Sept. 28, 1960	Design Personnel Procedure	Station shut down - safety test	Safety shutdown resulted when the 1C loop T_h receiver-indicator was calibrated above the 522°F shutdown setting.
Sept. 22, 1960	Personnel	Station operating - 56 MW	During preparations to charge resin into the 1BD purification loop demineralizer, a pocket watch from a technician's breast pocket fell into the demineralizer. The object was subsequently removed from the demineralizer.

Date	Basic Cause	Plant Operating Conditions	Remarks
Sept. 25, 1960	Material	Station start-up	Rod drop protection safety shutdown resulted at 51% power due to a loose Norwood T_{avg} 490°F micro-switch contact, which was closed at an average temperature of 497°F in lieu of being open above 490°F.
Dec. 7, 1960	Material	Station operating - 58 MW	Loop status safety shutdown resulted with three loops in service, when the 1D loop hydraulic valve relay (20LIOX-4) was energized by magnetic amplifier drift.
Jan. 27, 1961	Design Material	Station operating - 58 MW	Hydrostatic test of the 1A reactor coolant pump at 2750 psi revealed leakage around the periphery of the gasketed flange.
Jan. 28, 1961	Procedure	Station shut down - shutdown test	Safety shutdown during a special shutdown test due to high T_h following rapid load reduction (0.7 MW/sec.) was actuated in spite of rod insertion initiated at average temperature of 507°F and lifting of decay heat removal relief valve.
Jan. 30, 1961	Personnel	Station shut down - radiation survey	During performance of reactor head radiation survey three pairs of anti-c coveralls, which were placed on reactor vessel head fuel ports to protect the feet of survey personnel from high metal temperature, ignited causing damage to rod drive mechanism stator no. 53 thermocouple.

Date	Basic Cause	Plant Operating Conditions	Remarks
April 3, 1961 to April 5, 1961	Material	Station shut down - physics test	Approximately one-third of the condensate storage tank dome top collapsed when the vacuum breaker failed to operate when the tank was subjected to an unknown negative internal pressure.
April 4, 1961	Personnel	Station shut down - training	Approximately 1/2 of the diesel generators air intake louvers were damaged when a canvas cover was blown over the air intake while the diesel was operating.
April 5, 1961	Personnel	Station shut down - training	The 1A reactor coolant pump was operated in slow speed approximately 12 hours, without first venting, as required when the reactor coolant system pressure has been reduced below 200 psi.
April 6, 1961	Personnel Procedure	Station shut down - physics test	Misalignment of NIS recorder and intermediate range log level cause a recorder contact safety shutdown when the flux level was increased to 5 x 10^{-7} amps.
April 12, 1961	Design Material	Station operating - training	During the operation of the RWD incinerator abnormal amounts of smoke were observed issuing from the exhaust stack and loading hatch exhibiting a pressure build-up in the shell and hot spots on the wet gas scrubber.

Date	Basic Cause	Plant Operating Conditions	Remarks
April 19, 1961 to April 20, 1961	Material	Station operating - 60 MW	Safety insertions were initiated by the 4N series bistable mag-amps at approximate power levels of 106% and 104% respectively. Cause was drift in magamp set points.
May 2, 1961	Design	Station operating - 60 MW	Following repairs to the two air-operated charging valves, excessive vibration of the charging system piping accompanied opening of one charging valve. The cause of the vibration was determined to be leakage through the two check valves downstream of the charging valves.
May 17, 1961	Unknown	Station operating - 60 MW	A safety insertion occurred when the reactor power level was approximately 103.5%. Investigation of the magnetic amplifier saturation points showed them to be in the 109.5 - 111.5% range.
June 13, 1961	Personnel	Station shut down and cooldown	During the performance of the shutdown check list, the pressurizer pressure indicator-receiver was manually driven up-scale to determine the low-pressure protection set point. The pilot-operated relief valve opened at 2175 psi and the solid primary system pressure dropped to approximately 100 psi with the reactor coolant pumps in slow speed.

Date	Basic Cause	Plant Operating Conditions	Remarks
June 15, 1961	Material	Station shut down - physics test	A defective CIC cable connection resulted in a high start-up rate safety insertion during a physics test when the reactor was subcritical and the control rods were being manually inserted.
June 21, 1961	Design Procedure Personnel	Station start-up - FEDAL test	A safety shutdown resulted at 22% power and was initiated by the 5N series magnetic amplifiers remaining in the 24% biased start-up protection position when the source range selector switch was not placed completely in the pullout position to remove the 24% protection.
June 22, 1961	Material Personnel Procedure	Station start-up - FEDAL test	A safety shutdown resulted at 22% power and was initiated by the 5N series magnetic amplifiers remaining in the 24% biased start-up protection position when the source range selector switch was not placed in the pullout position.
July 19, 1961	Personnel	Station shut down - physics test	A safety shutdown was actuated by the low-pressure protection contact scram. The cause was determined as a feedback signal from the wide range pressurizer recorder, which was being repaired, to the wide range Norwood pressurizer pressure receiver indicator.

Date	Basic Cause	Plant Operating Conditions	Remarks
July 22, 1961	Material	Station shut down - physics test	A safety shutdown resulted from a spurious full-scale deflection signal to the NIS Channel B recorder resulting in a shutdown at 5 x 10^{-7} amps.
July 27, 1961	Material	Station operating - 20 MW	A safety insertion occurred due to a loose contact in the pump speed circuit resulting in false flow information to the reactor protection system.

CHAPTER 5

RADIATION EXPOSURE AND CONTAMINATION CONTROL

Introduction

As a result of strict adherence to health physics procedures, the station had a very satisfactory radiation safety record during Seed 2 lifetime. Radiation dosages fluctuated with the maintenance work load in radiation areas, but were generally much lower than those allowed by applicable regulations. As was expected, refueling operations proved to be the largest single contributor to personnel exposure.

The number of openings of the primary system and its auxiliary systems for maintenance increased over those experienced in Core I Seed 1, thus creating more sources of contamination and, subsequently, more problems of contamination control. No incidents occurred that could be classified as major, i.e., an incident requiring immediate extensive decontamination of a working area. There were several minor incidents which involved the spread of contamination from contaminated areas to controlled areas. These controlled areas are designated as boundaries to the contaminated areas, and consequently are surveyed frequently to assure positive control of contamination.

Radiation Intensities

While gradual increases in radiation levels were being measured in the reactor coolant system, no increase in the normal background radiation occurred in the office spaces and occupied areas of the plant. Radiation levels in the access areas of the reactor plant have remained essentially the same as those measured during Core I Seed 1. Figure I-3 shows these levels measured on July 21, 1961, near the end of Seed 2 life.

Radiation levels measured on the operating loops at 100 percent power are shown in Figure I-4. There was no significant change in these intensities from Core I Seed 1, although a point of higher intensity (40 R/hr) was found near the coolant line on the inlet side of the B heat exchanger. This is the maximum intensity ever measured in the compartments. In order to minimize personnel exposure surveys at power are necessarily cursory.

Average radiation intensities in the reactor head area have remained nearly the same, with only slight increases being detected. The highest radiation level detected in conjunction with maintenance work was 1.6 R/hr on contact with the reactor head. More data are presented in Part V, Chapter 3, Radiation Level Build-up Experience on Components and Piping.

Radiation Intensities Measured During Full-Power Operation with Reactor Pit Drained

On April 15, 1961, the reactor was operated at 95 percent power with the reactor pit drained. This is not a normal station operating condition, but repair to the reactor pit protective coating had been made and the coating was being cured. Small increases in background radiation were measured in various sections of the reactor service building. In the alleyway between the radiochemistry

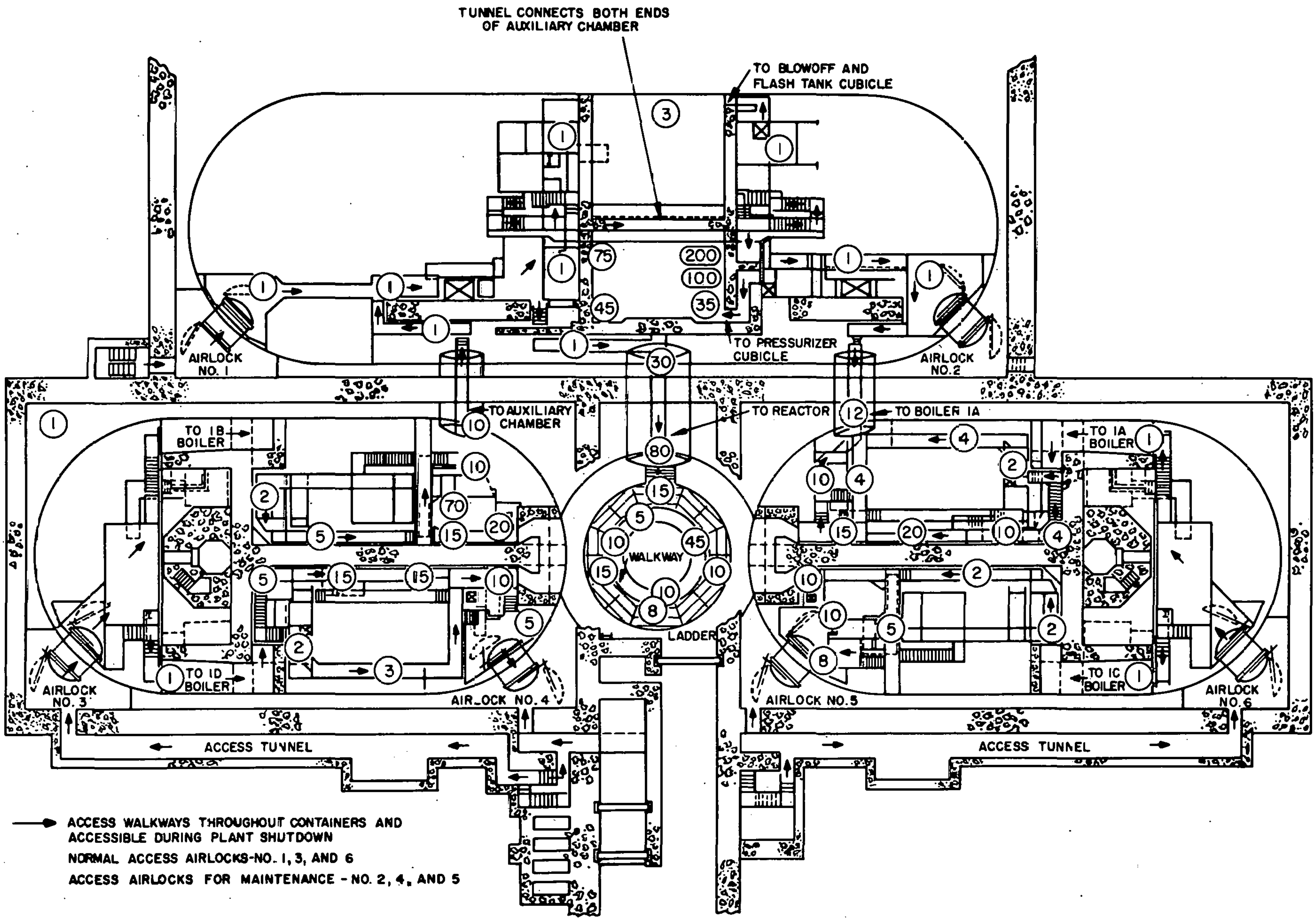

Figure I-3. Radiation Levels (gamma mr/hr) at Zero Percent Power.
(3 Days After Shutdown - August 1961)

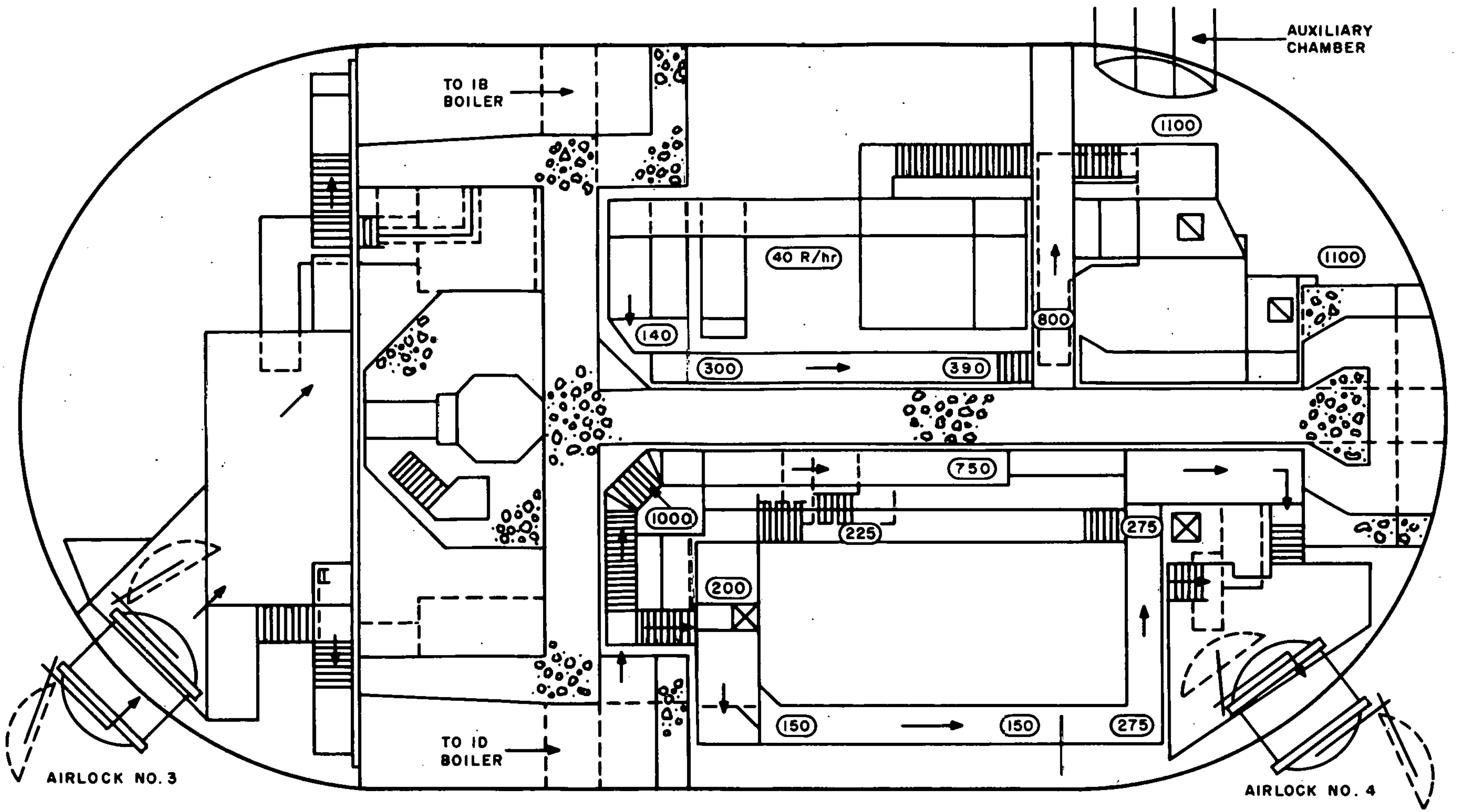

Figure I-4. Radiation Levels (gamma mr/hr) at Full Power.
BD Boiler Chamber

laboratory and the 1B-auxiliary equipment room, the average increase in radiation level was 0.05 mr/hr above background. In the test and training building, average increases were found to be 0.03 mr/hr above background. No neutron radiation was detected in any of the areas surveyed outside of the fuel handling building. In the fuel handling building, significant increases in the radiation levels, both neutron and gamma, were confined to the reactor pit and those areas adjacent to it, the canal walkways, and the east and west balconies. On the canal walkways next to the pit, the radiation level averaged 2 mr/hr; normal for this area, with water in the reactor pit, is 0.04 mr/hr. Above the reactor pit and parallel with the walkways the average radiation level was 9 mr/hr and the average neutron radiation was 1400 thermal neutrons per second per square centimeter. Figure I-5 shows gamma radiation and neutron radiation measured at various depths in the reactor pit.

Radiation Exposure Control

The "exclusion area" procedure, whereby personnel are admitted to potentially high radiation areas only under strict control, has been continued with the same favorable results in preventing accidental high exposure during power operation. This control requires only a few rigidly enforced rules. Most radiation exposures are received in doing maintenance and testing on shut down components. Consequently, most of the radiation safety effort must be directed toward reducing exposures from this source.

The most significant expansion of the radiation safety program has been in the areas of training and distribution of printed material. Several hours of training have been provided each individual in the form of lectures and safety meetings. In addition to the lectures, the Duquesne Light Company has published and distributed to personnel throughout the organization a booklet on radiation safety at Shippingport. A basic explanation is provided of the types of radiation to be found at the station and some of the properties associated with them. A second pamphlet has been purchased and distributed to all station personnel describing, in general, the various properties of radiation and associated problems. It is still necessary, at times, to provide continuous radiation safety guidance during certain maintenance operations. Usually this guidance is provided when penetration of any portion of the primary system or work in the area of the reactor head is involved. The radiation technician assigned to cover such work provides information as to radiation levels, personnel protective equipment required, estimates of time allowed in the work area, need for temporary shielding, and general radiation safety rules that should be followed.

Contamination Control

The problem of contamination control at Shippingport has evolved into a very definite pattern. First, those areas of the plant where contamination is most likely to occur are plainly marked with contamination warning signs at all entrances and exists. Rules concerning protective clothing and tool and equipment control are explained to all persons who might have access to the area. Frequent surveys of the area are made to determine if excessive contamination is building up or if contamination that is present is becoming an airborne hazard; decontamination is initiated if contamination is excessive or an airborne hazard develops.

Areas of controlled access border the contaminated areas and include the reactor plant contaminated locker room. These areas are surveyed frequently and generally remain free of contamination. Several times contamination has spread to these areas but has been detected and

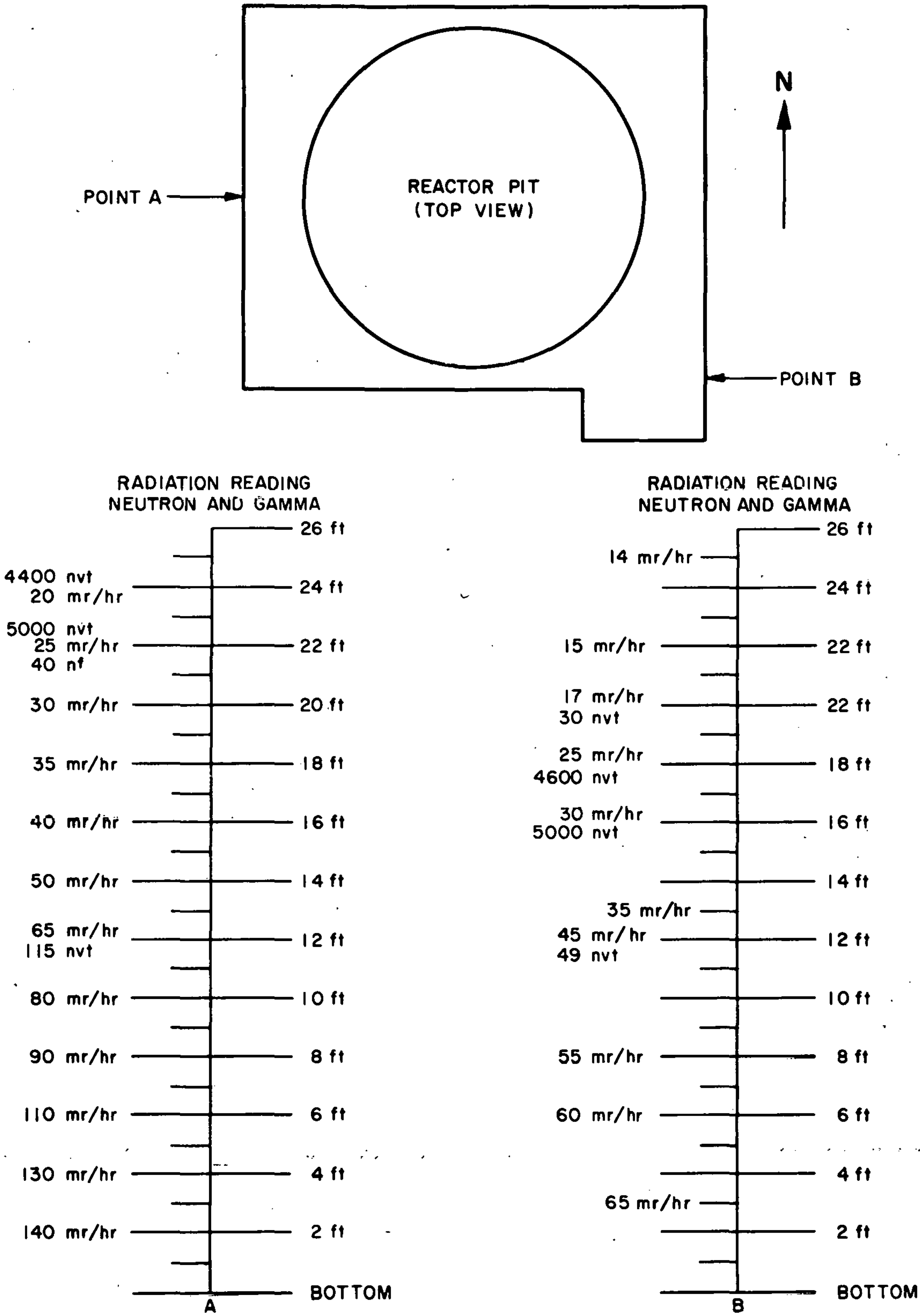

Figure I-5. Radiation Levels Measured in Reactor Pit with Pit Drained.

quickly removed to prevent spread to the remaining areas of the plant which are designated as clean areas. These areas are surveyed periodically to assure that clean limits are observed.

To prevent the spread of contamination from one area to another, specific rules have been established regarding the flow of men and materials in and out of a contaminated work area. All work in contaminated areas is closely observed by Health Physics and anti-C clothing is provided for those persons doing the work. All tools and other items being removed from the work area are treated as contaminated until proved otherwise. If contamination is present, the item is cleaned in the decontamination room before re-issue. Personnel leaving the area proceed directly to the contaminated locker room where monitoring facilities are available in the form of "friskers." Other monitoring stations are located between the clean and contaminated locker rooms, and if contamination is detected, a shower is mandatory. After removing all anti-C clothing, the person goes to the clean locker room. After the person is dressed, he then must pass a hand and foot monitor and a portal monitor before he can leave the building. An additional portal monitor is provided at the exit from the site. Figure I-6 shows a floor plan of the Reactor Service Building with significant monitoring stations and control points pointed out.

An additional room acquired by the Health Physics Group since Core I Seed 2 refueling has proved to be an excellent control point for preventing the spread of contamination and contaminated items. This room, called the Health Physics Field Office, is located conveniently to the clean and contaminated locker rooms, the laundry, the canal area, and the decontamination room. Personnel entering the reactor plant containers must check with the field office before going to their work area. This provides Health Physics with a record of all jobs in progress and allows the radiation technician to provide instructions to maintenance and construction personnel concerning radiation and contamination control. An additional automatic smear counter has been located in this office to expedite contamination checks. One radiation technician is assigned to this area to provide a continuous control of all Health Physics regulations.

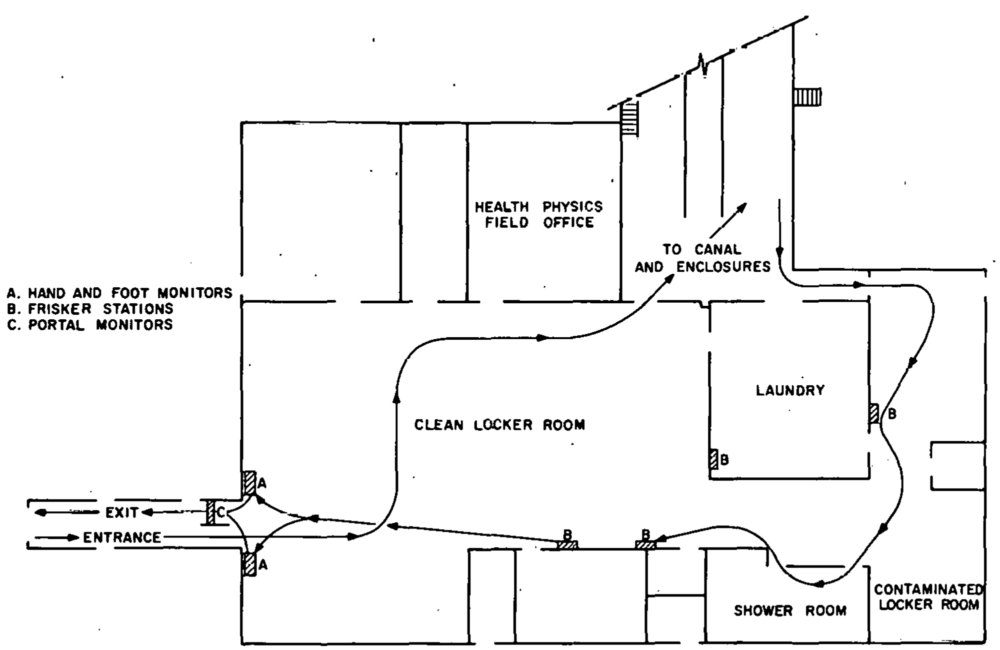

Figure I-6. Floor Plan of Reactor Service Building.

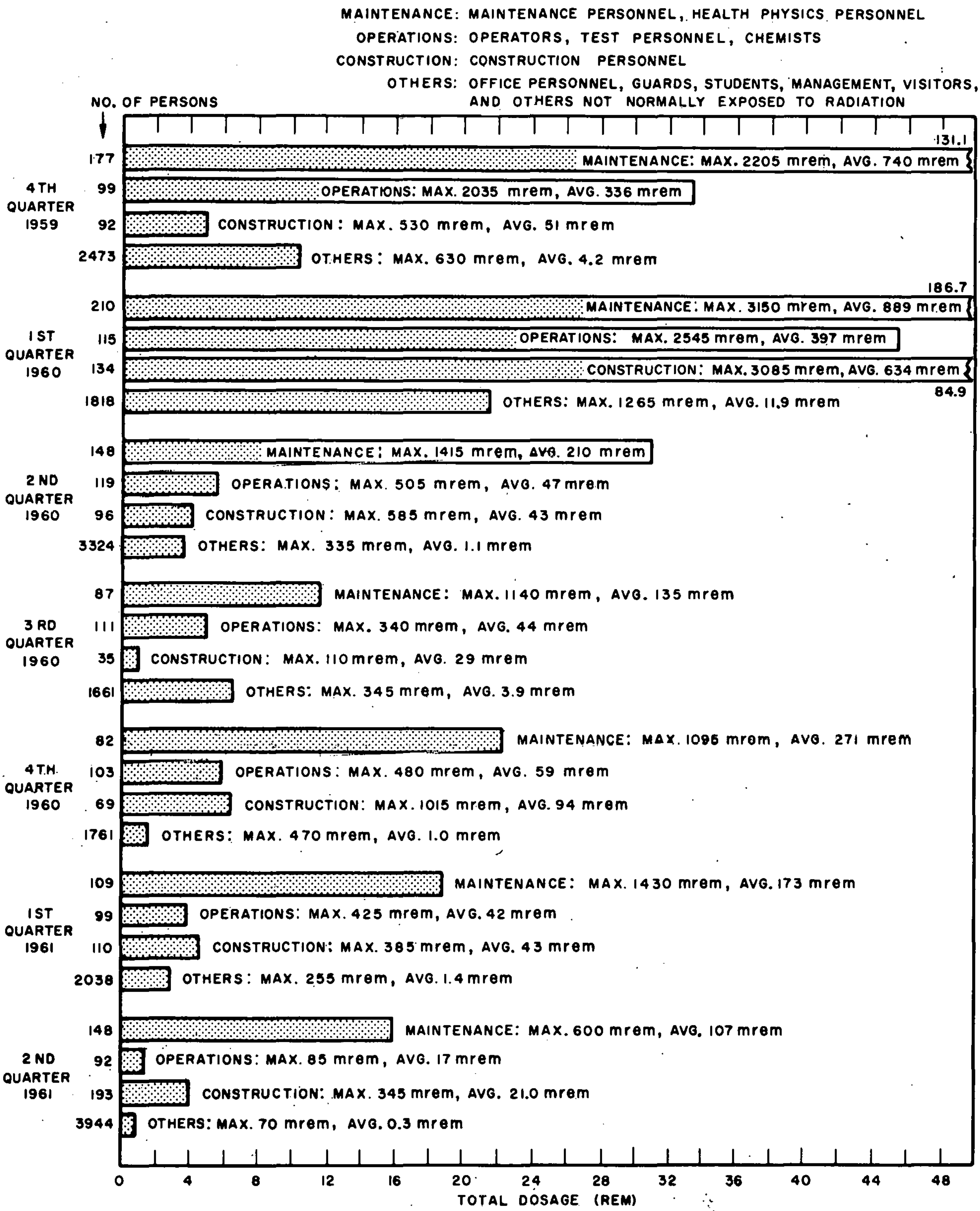

Figure I-7. Shippingport Atomic Power Station Radiation Dosage Summary.

Contamination of Refueling Building

In June, 1961, the canal area was surveyed to ascertain the cause of frequent contamination occurrences on the canal walkways. The survey revealed that contamination levels, which averaged 1500 dpm/ft^2 (680 $\mu\mu$c/ft^2) existed on piping, conduit, and flat surface ledges high on the canal walls. The contamination was in the form of a fine dust and was easily transferred. It is felt that much of the walkway contamination resulted from this dust, but it is not known how the original contamination build-up occurred. The area was vacuum cleaned and the contamination level was reduced to an average of 400 dpm/ft^2 (180 $\mu\mu$c/ft^2).

Radiation Dosage Experience

Figure I-7 shows the total dosage received by personnel in different work groups. Maintenance personnel received the most radiation dosages and most of their dosage was received during the refueling period. The average dose to maintenance personnel for the entire period was about 12 percent of the 3000 mrem allowed under the National Committee on Radiation Protection recommendations. Most of the exposure received during non-refueling periods was received in 5-20 mr/hr radiation fields.

Emergency Planning

The Duquesne Light Company has organized and initiated an emergency plan with the Aliquippa Hospital to provide care for persons who may be injured while working in contaminated areas. Approximately twenty hours of instruction and demonstration have been given to the staff at the hospital and a test drill was performed as a check on the effectiveness of the plan. Briefly, the plan outlines the steps to be taken by the hospital, when notified a potentially contaminated person is being brought to the hospital, and the steps to be taken by Duquesne Light Company to prevent the spread of contamination from the pre-arranged control areas of the hospital and to assure that all contamination is removed.

Other emergency procedures have been formulated defining emergency conditions that could occur at the station. These are broken down into minor release to the atmosphere, major release to the atmosphere, and uncontrolled release to the river. A test drill was performed of the minor release to the atmosphere; future drills are planned to simulate each of the emergency conditions.

Summary

Radiation intensities measured in the primary plant have shown gradual but definite increases throughout the life of Seed 2. The maximum measured radiation intensity (40 r/hr) was measured on an operating loop which is not a normal access area while the station is at power. The average radiation level in the compartments after shutdown was approximately 15-20 mr/hr. No information is available concerning radiation intensities in the reactor chamber while at power. However, radiation intensities, both neutron and gamma, were measured in the reactor pit and surrounding areas while the reactor was at power and the water shield removed. While the radiation in these areas was measurable, it did not prohibit access.

Contamination control in the primary plant has become more of a problem due to the increase in maintenance on the primary system. Several changes, including building modifications, have been made to assure positive control of personnel working in contaminated areas. A health physics work center is located at the entrance to the reactor plant to facilitate giving radiation safety information and monitoring tools and equipment leaving the area.

Personnel radiation exposures were generally low, except during the refueling period. In all cases exposures have been in compliance with Federal and State regulations.

Emergency plans have been prepared to cover all emergency conditions that could occur at the station. Some test drills have been performed and future tests are planned to simulate each of the emergency plans.

CHAPTER 6

MAINTENANCE

Introduction

A more effective maintenance program was conducted during Core I Seed 2 operation due to experience gained in the technology associated with maintaining a nuclear power station. Maintenance work was performed on practically all types of reactor and turbine-generator plant equipment, from instrument calibration and feedwater heater repairs to the removal and installation of a hermetically sealed main coolant pump and volute and the replacement of a mechanism position indicator coil housing assembly and stator-water jacket assembly.

The effectiveness of the maintenance program contributed to the station capability of maintaining a very high availability for testing, power production, and training throughout the entire Seed 2 operation.

Manpower Requirements

Figure I-8 presents data showing the total manhours per month worked by the maintenance group. During operation of Core I Seed 2 the number of "in-station" maintenance personnel remained the same (approximately 60) as it was at the end of Core I Seed 1 life. When the work load became too great for the "in-station" maintenance personnel to handle, particularly during the two months preceding Seed 2 - Seed 3 refueling, maintenance personnel from other power stations of the Duquesne Light Company were brought in to alleviate the work load. The increase in manhours several months prior to Seed 2 - Seed 3 refueling was due primarily to the preliminary preparations and personnel training required for the subsequent refueling operation.

Follow-Up of Major Maintenance Jobs During Seed I Operation

During Core I Seed 1 operation, components in the reactor and generator plants presented several major maintenance problems, which were successfully resolved and created no further problem for Seed 2 operation. Problems were encountered in the rod drive inverters, moisture separator, the turbine, RWD hydrogen burner, and the acid system.

1. Rod Drive Inverters

The rod drive mechanisms receive power from low-frequency mechanical inverters. During Seed 1 life, the inverter faceplates were replaced with a type constructed of glass melamine phenolic sheets to overcome a creep problem caused by the original paper base phenolic plates. The new plates were trouble-free during Seed 2 life. The original brushes for these inverters were made of 62 percent copper, 27 percent graphite, and 11 percent molybdenum disulfide; they were replaced during Seed 1 operation with brushes made of 75 percent copper, 25 percent graphite. This change was made to correct the problem of brush material plating out on the commutator segments. However, these replacement

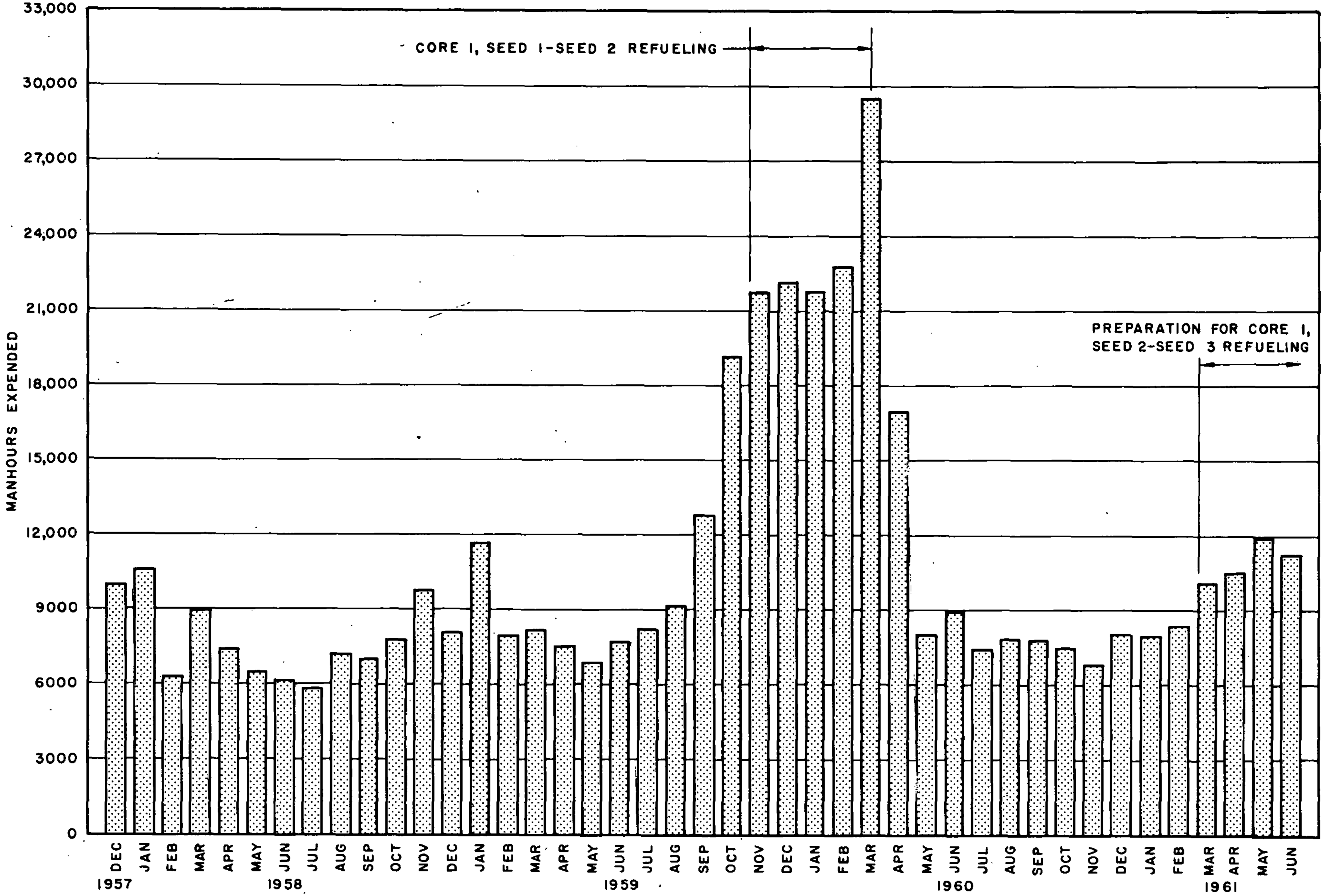

Figure I-8. Station Maintenance Monthly Manhours, December 1957 to June 1961.

brushes were subjected to oxidation and were, therefore, unsatisfactory. A third brush material containing 75 percent silver was tried during Seed 2 operation and proved to be successful. These changes reduced maintenance on the rod drive inverters to two cleanings in 10,000 hours of operation during Seed 2 life, a reduction of 80 percent in maintenance time.

2. Moisture Separator

The moisture separator, which had completely failed during Seed 1 operation, was operated throughout Core I Seed 2 life without incident or failure indication. The design defects present in the original model of the tuyere were eliminated. The moisture separator was inspected on June 15, July 17, and November 29, 1960 and April 3, 1961; all internals were found in good condition. The new tuyere design consisted of changing blading from a flat to a curved blade, from type 304 stainless steel to 12 percent chrome steel, and from 1/8 inch to 3/16 and 1/4 inch thickness. The tuyere, originally a single unit, was redesigned and now consists of four flanged sections which are bolted together. The blades were welded in place by establishing a preheat of 300°F. Upon completion of welding each section was then stress relieved at 1175°F for 12 hours and furnace cooled.

3. Turbine

The turbine operated during Core I Seed 2 life without incident or failure indication. The last row of blading was inspected on December 2, 1960 and found in reasonably good condition.

4. RWD Hydrogen Burner

In May, 1960 a new design hydrogen burner was installed in parallel with the existing hydrogen burner in the radioactive waste disposal plant for trial operation. Following successful trial operations, the new burner was placed in service and has since operated trouble-free. The old burner was placed in a standby condition. The new type burner is superior because the catalyst can be rejuvenated in place by purging with nitrogen to remove the moisture from the catalyst. With the old hydrogen burner, the burner had to be isolated, the catalyst removed, and rejuvenated by acid cleaning. Even after rejuvenation, the old catalyst would operate satisfactorily for only about one hour before failing.

5. Acid Handling

Prior to start-up of Core I Seed 2, the ball-check piston-type acid pump was replaced with a diaphragm-type acid pump. This pump is used to regenerate the demineralizers in the Turbine Generator Plant. Considerable maintenance was necessary on the piston-type pump during Core I Seed 1 operations. The diaphragm-type acid pump has operated satisfactorily during Core I Seed 2 life. The demineralizer stainless-steel acid inlet line spool-piece to the acid eductor was replaced with a polyvinyl plastic pipe to eliminate acid leakage and to improve the acid mixture concentration for the demineralizer regenerations.

Major Maintenance Items During Seed 2 Life

1. Mechanism Stator Replacement

Shortly after Core I Seed 2 was placed in operation, the E-12 rod drive mechanism stator winding indicated a very low value of resistance to ground. During a shutdown all rods were de-energized except E-12, which was energized through the rod test circuit in an attempt to bake the winding and remove the suspected moisture. The baking operation successfully increased the insulation resistance for a short period of time, but did not provide a permanent solution to the problem. It was, therefore, necessary to replace the defective stator with a spare unit. The reactor pit was drained. The reactor chamber dome, ventilation ducts, ladders over the rod drive mechanism, and control rod holddown structure were removed. The E-12 position indicator coil housing assembly and the mechanism stator assembly were removed from the reactor vessel head as a unit, and a spare stator and position indicator coil housing assembly were installed as a unit. No penetration into the reactor coolant system was required for this operation. All electrical checks were performed successfully on the replacement stator and position indicator coils. The holddown structure, ladders, ventilating ducts, and the reactor chamber dome were re-installed to complete the operation. Total elapsed time for this operation was approximately 40 hours.

2. Radioactive Waste Disposal System Incinerator

The RWD incinerator and gas scrubber unit has operated unsatisfactorily during the life of Seed 2. Its operation was uncontrollable and caused minor furnace explosions, overheating, and release of a great deal of smoke to the atmosphere. The first maintenance work on this unit was devoted to improving the filter. The original filter design was modified to permit a more efficient exchange of the filter elements. Angle supports were welded in the bottom of the filter section to support the modified filter, and a plate was bolted to seal the opening where the filter is installed. The modified filter could then be replaced by removing the plate, sliding out the filter assembly, replacing the filters in the filter assembly, reinstalling the filter assembly, and resealing the opening with the plate.

In June of 1961, the gas scrubber was removed from service, dismantled, and inspected. The internals were coated with approximately 1/4 inch of carbon and, upon cleaning, were found badly corroded. Due to overheating, one portion of the sidewall had pulled away from the baffle plates. A piece of angle iron was then installed and the baffle and sidewall were rewelded. In the blower assembly, as much as to 2 inches of carbon sludge was found coating the impeller and housing. In the filter shell and inlet piping, numerous holes were found and welded closed. Carbon sludge deposits of 1/2 inch were also cleaned out from both inlet and outlet sides of the filter. All sludge and carbon was tested and found to be contaminated. A draft gage and damper are to be installed on the discharge side of the blower unit in order to obtain better control of the burning. It is believed that this modification and the scrubber repairs will improve operation and decrease maintenance.

3. <u>1A Feedwater Heater Tube Replacement</u>

Leak testing of the secondary fluid systems in April, 1960 revealed that 52 additional tubes had failed in the first pass and 5 in the second pass of the 1A feedwater heater since the previous test in January, 1960. This represented a total of 78 tube failures in the first pass and 5 tube failures in the second pass. All tubes of the first pass were plugged, a hole was cut in the division plate between the first and second passes and the heater was operated as a single pass unit. The heater was then placed in service May 6, 1960. Leak tests performed in the early part of July, 1960 revealed that the 1A feedwater heater was again leaking severely. On July 18, it was removed from service, isolated, and inspected.

Several ruptured tubes were removed and sent to the heater manufacturer, Duquesne Light Company Chemical Lab, and a government laboratory for further analysis. The results of these investigations revealed that (1) stress corrosion was the cause of the tube failures and that (2) the original manufactured tubes were not stress relieved at the bends. Based on these findings, it was recommended that the heater be completely retubed.

The shell internals of the 1A heater were inspected in August, 1960; some pitting was found. The inside surface of the shell was wire brushed and coated with Apexior to prevent further pitting. Removal of the old tubes began in October, 1960, and was completed by November 17, 1960. The tube sheet was sent out to the manufacturer for removal of the tube stubs; it was returned on December 14 and tube installation was started.

Retubing was completed on January 17, 1961, and the heater was returned to service on February 1, 1961. Due to the work load in other areas of the plant, this work was performed on a part-time basis. The original tubes were Admirality metal type B; the newly installed tubes were Arsenical-Admirality, metal heat-treated for stress relief. Since the installation of the new tubes, the 1A feedwater heater has performed satisfactorily.

4. <u>1A Heat Exchanger Leak Testing</u>

The 1A reactor coolant loop was removed from service in December, 1960, drained, and flushed in preparation for leak testing the 1A heat exchanger. The test method required pressurization of the secondary side of the heat exchanger to 75 psig with control air, filling the primary side with water up to the elevation of the top handholes, and observation of the tube sheet for bubbles. When no leaks were found at 75 psig, the pressure was increased to 150 psig using bottled N_2 gas. In order to observe the top five rows of tubes, a periscope was constructed and installed in one of the handholes. The top handhole openings were then sealed so that the water level could be raised to cover the top five rows of tubes. No leaking tubes were found. Following this testing, the handhole covers were welded in place and the loop was hydrostatically tested at 2750 psig and returned to service.

5. <u>12-Inch Main Steam Stop Valve Disc and Seat Failures</u>

The 1A motor-operated (40-H2-1), 1A manual (40-H6-1), 1C manual (40-H6-3), and the 1D manual (40-H6-4) main steam stop valves were dismantled for inspection and replacement of the bonnet seal rings between April 2, 1960 and April 7, 1960. The 1D motor-operated steam stop valve (40-H2-4) was taken out of service on November 21, 1960 and disassembled for repair of a leaking bonnet seal ring. Each valve has a nominal pipe size of 12 inches and is rated at 1050 psig maximum pressure and 600 psig nominal rating.

The condition found in each valve during inspections was as follows:

The 1A Motor-operated (40-H2-1) valve had a steam cut approximately 3/4 inch long in the body of the valve next to the seal ring. A quarter-inch spacer was machined and installed above the new seal ring. This spacer repositioned the seal ring below the cut in the body and allowed the ring to seal on the machined surface that had not been damaged. The wedge and valve body seats were in good condition. The seat rings (downstream) on the 1A manual (40-H6-1) valve were found to be flat on the top portion. Approximately four inches of the top section of this ring was separated 3/8 inch from the body and prevented the downstream side of the wedge from sealing. Inspection of this valve by the valve manufacturing representative failed to determine the exact cause of the seat ring deformation. A new pressure seal ring was installed and the valve was reassembled.

A similar condition of the seat rings was found on the 1C Manual (40-H6-3) valve but not as severe as in the 1A. Measurements were taken to determine the exact amount of deformation of the seat rings. The amount of deformation on the upstream side of the valve was 0.091 inches and on the downstream side of the valve it was 0.075 inches. A new pressure seal ring was installed in the bonnet and the valve was reassembled.

The seat rings on the 1D Manual (40-H6-4) valve were in the same condition as those on the 1A and 1C valves. Measurements showed that deformation on the upstream side was 0.150 inch and on the downstream side it was 0.197 inch. A new pressure seal ring was installed in the bonnet and the valve was reassembled. Just prior to reassembly of this valve, a gouge was discovered on the bonnet seating surface, apparently caused by a punch when the split rings were being removed. The bonnet was sent to the machine shop and approximately 0.015 inch was machined from this surface.

In September, 1960 the 1D motor-operated steam stop valve (40-H2-4) started to leak severely at both the bonnet and packing in the closed position. In November, 1960 the valve was removed from service during station cooldown and disassembled for repair of the leaking bonnet seal ring and the packing. Hairline cracks were found on both faces of the stellite valve disc. These faces were cleaned and the disc was reinstalled in the valve. The valve pressure seal ring was cut in some areas of its knife edge seating surface. A new pressure seal ring was installed in the valve. Numerous axial grooves were scored on the side of the valve stem at the packing gland area. These were approximately 0.1 inch deep and 0.2 inch wide in the worst case. Inspection of seat rings indicated that they were in good condition. It was noted at this time that the rings were not seal welded to the body as suspected by the manufacturer. The internal surfaces of the valve were in good condition, except for one hairline cut in the body at the seating surface of the pressure seal ring. This cut was removed by feathering the metal on both sides of the cut. All surfaces of the valve were cleaned, reassembled, and fitted with new packing. The valve was then returned to service.

It is planned to remove the 1A manual (40-H6-1), 1C manual (40-H6-3), and 1D manual (40-H6-4) main steam stop valves from the steam system during Seed 2 - Seed 3 refueling. These three valves are to be returned to vendor for reconditioning. The 1A, 1B, 1C, and 1D (40-H2-1, -2, -3, and -4) motor-operated and the 1B manual (40-H6-1) main steam sto valves are to be dismantled, inspected, and repaired in place, if necessary, during Core I Seed 2 - Seed 3 refueling.

6. Reactor Coolant Pump Flange Gasket Leakage

During the leak test of the 1B heat exchanger in April, 1960 leakage was detected at the volute gasket on the 1B reactor coolant pump. After isolation and draining of the loop, four flange bolts, 90 degress apart, were measured for installed stretch condition. These bolts were heated, removed, and measured after they had been cooled to ambient temperature. Measurements indicated that these bolts ranged from 5 to 7 mils under the prescribed elongation. The bolts were reinstalled and stretched to the proper torque. The remaining 20 bolts were then heated and stretched the additional 5 to 7 mils for proper torque. The pump was then successfully leak tested, but at this point an investigation was made to determine what the maximum bolt loading on the pump flange would be if one or more of the 20 bolts, which were assumed to be 5 to 7 mils under the prescribed elongation, were less than this amount prior to reheating. Calculations indicated the possibility of flange distortion and bolt damage under the worst conditions. It was then decided to remove all bolts, install a new gasket, and restretch all bolts to the proper torque. The flange gasket was then leak tested at 2750 psig and no leakage was found.

During the leak test of the 1A heat exchanger in February, 1961 leakage was detected at the 1A reactor coolant pump flange gasket. All the bolts around the flange were measured as found. Then four bolts at a time were heated, removed, allowed to cool, and measured. After the bolts were reinstalled in the flange, they were heated and stretched to the desired 0.014 ± 0.001 inch. All bolts were again measured and found within permissible stretch limits. The pump was then leak tested successfully.

7. Reconditioning and Modification of the 1D Reactor Coolant Pump Volute

Prior to decontaminating the reactor coolant pump volute in preparation for reconditioning and modification in conjunction with pump replacement, contact radiation readings were taken. After several scrubbings of the volute with a solution of EDTA and alconox, the radiation levels were reduced approximately 50 percent. Additional decontamination was performed by soaking the pump volute in a steam heated solution of alconox and EDTA. After three soak and rinse operations, it was evident that further decontamination was not practical. Listed below are the radiation readings before and after each phase of the operation.

	Prior to Decontamination	After Scrubbing Operations	After Soaking Operations
Volute top flange	175 mr/hr	80 mr/hr	50 mr/hr
Volute outlet	70 mr/hr	50 mr/hr	50 mr/hr
Volute inlet	35 mr/hr	30 mr/hr	25 mr/hr

The inlet and outlet nozzles of the pump volute were then machined in preparation for welding the volute into the 1D loop. The inlet end (suction) was machined using a special portable 18-inch pipe beveling machine, but the outlet end (discharge) had to be machined on a horizontal boring mill since it was impossible to install the portable beveling machine at this location.

The second phase of the volute reconditioning involved machining the internal bore of the volute at the thermal barrier area and cutting back the water tip vanes to permit future installation of a larger pump and impeller for Core II operation. This was accomplished by means of a special portable boring machine which was designed and built for this operation. Figure I-9 shows the portable boring machine mounted on the pump volute.

At the completion of the boring operation, a dye check of the volute casting was performed. Numerous indications of flaws appeared in both the inside and outside of the volute. One crack in the shroud ring was ground to a depth of 1-1/8 inches before the crack disappeared. The ground area of the shroud ring was filled with weld material to the contour of the shroud ring and successfully dye checked. Most of the flaws were removed by grinding the surface area. Two areas inside the volute were filled with weld material, ground, and dye checked. These two areas were ground out to a depth of 1/2 inch to remove the flaws.

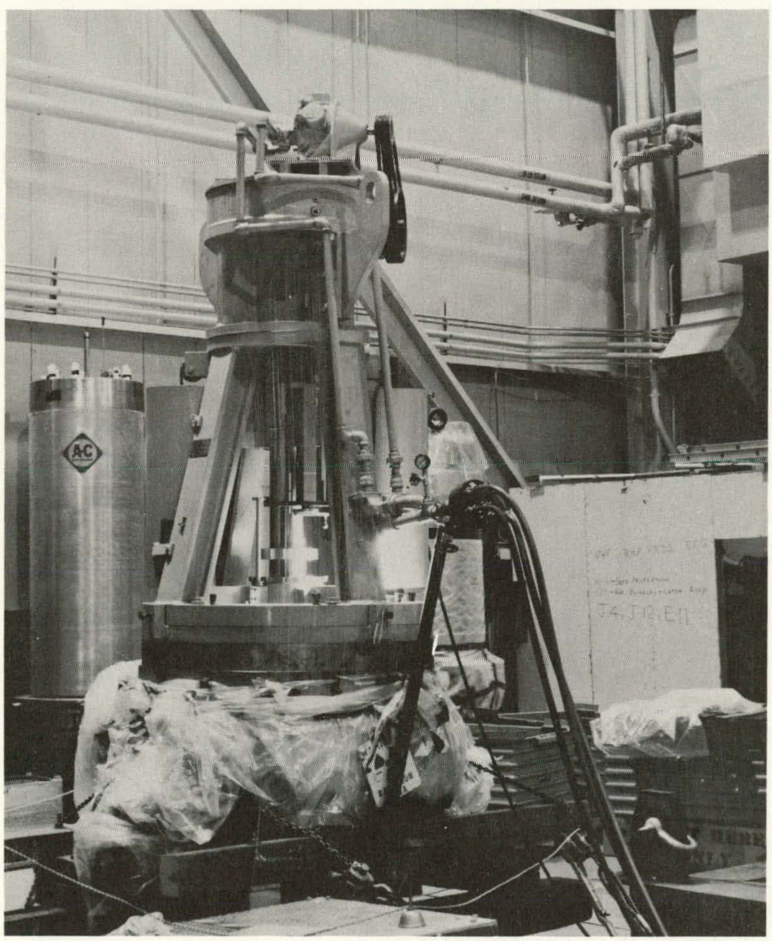

Figure I-9. Portable Boring Machine Mounted on the Pump Volute.

There were approximately twenty areas on the inside of the volute where flaws were removed by grinding to a depth of 1/8 to 1/4 inch. These areas did not require addition of weld metal; however, the sharp edges of the grooves were smoothed out to blend in with the contour of the volute. There were two areas on the outside surface of the volute where deep grooves were made from 3/16 to 1/2 inch deep to remove the flaws. The sharp edges of the groove were smoothed out to blend in with the contour of the outside of the volute.

8. Removal of Resin Sample from AC Purification Loop Demineralizer

During June, 1960 the 1AC demineralizer was removed from service to extract a resin sample so that a detailed analysis could be performed to obtain the following information:

a. Total long-lived activity removed by the resin.
b. Distribution of the activities on the resin between crud and absorbed ions.
c. Distribution of activities and chemical elements on the length of the bed.
d. Effect of resin in reducing plant contamination.
e. Radiation damage to the resin.

To obtain this sample, it was first necessary to remove the metal cover and concrete cover sections from atop the AC purification loop access chamber port. The bolts were then removed from the AC purification loop access port and the dome was removed. The concrete plug in the demineralizer shield cubicle was removed to provide access to the demineralizers. The top of the demineralizer was shielded with lead and a work platform was installed. The three-inch resin fill line was cut (using a standard pipe cutter modified to hold oversize cutting wheels) just above the demineralizer inlet fill nozzle. A special leak shielding cask was placed over the inlet fill nozzle and a probe was inserted through the cask. The probe was driven down into the resin bed until it contacted the top of the demineralizer outlet filter. The sample was then withdrawn into the cask. The concrete plug, chamber dome, and concrete hatches were reinstalled after welding the three-inch pipe and removing the platform and shielding. Upon completion of the installation of the chamber dome, the AC purification loop was returned to service.

9. Collapse of Condensate Storage Tank Top Section

In April, 1961 it was discovered that approximately 1/3 of the top of the condensate storage tank was "dished-in", apparently because of a vacuum being placed on the tank due to failure of the vacuum breaker. An inspection for failure of the tank was made with dye penetrant; no cracks were found and the top was assumed to be sound. Pieces of angle iron were welded to the dished-in portion of the top and, with the use of the crane boom and 2 chain falls, the top was successfully pulled out. The overflow pipe hanger which had buckled was replaced.

The level indicator was found to have a ruptured diaphragm. The diaphragm was subsequently replaced and the instrument calibrated. The vacuum breaker on the tank was removed and inspected. It was found that the vacuum breaker shaft could stick and remain closed. The shaft was cleaned, reset, and reinstalled. A periodic check of the vacuum breaker has been initiated to insure proper operation.

The inside of the tank was inspected and found to have a slight pitting in the surface. The inside of the tank was then wire brushed and painted with Apexior No. 1 to prevent further corrosion. The tank was successfully leak tested and returned to service.

10. Primary System Valves

Minimum maintenance problems were experienced on the reactor plant stainless-steel valves, with most of the valves operating trouble-free during Core I Seed 2 operation. The few exceptions are discussed in the following paragraphs.

From December, 1960 to August, 1961 leak rate tests performed on the two pressurizer steam relief valves (06-H15-1 and 10) indicated an average leakage of 31.6 gal/hr from 06-H15-1 and 25.3 gal/hr from 06-H15-10.

During the same period the leak rate for the four reactor water relief valves (06-H15-2, 3, 8 and 9) averaged 4.0 gal/hr, 1.2 gal/hr, 8.4 gal/hr, and 0.7 gal/hr, respectively. It is planned to dismantle the numbers 1, 2, 8, and 10 valves for inspection and repair during Seed 2 - Seed 3 refueling.

In May, 1960 in preparation for initial start-up of Core I Seed 2 operation, the FEDAL crossover isolation valve stem was reported to be galled in approximately the three-quarter open position and could not be operated in either direction. The valve was removed from the system to permit dismantling, inspection, and repair. Since there are 22 such valves presently installed in the reactor plant, it was decided to investigate thoroughly this particular valve for the cause of galling.

Upon dismantling and inspection, it was found that the lower threaded portion of the valve stem had been galled to the disc threads. All clearance and internal mating parts were checked for any discrepancy from the specified dimension which might have contributed to the failure. From the measurements taken, there appeared to be sufficient clearance between mating parts for normal operation of the valve. Unfortunately, no accurate measurement of the threaded parts could be taken because of the nature of the failure. The remaining threads on the bottom portion of the stem were not in good enough condition to be measured accurately with a thread micrometer for comparison to the new stem threads.

One item, which may be of significance in connection with the threaded parts in question, is a revision noted on the supplier's drawing. This revision notes that the lower threaded portion of the valve stem had been modified from a Class 2 to a Class 1 fit. This change raises the question that there may have been some difficulty of a similar nature during initial manufacturing and testing. Since it is possible that there are other valves of this type in operation in the plant, which may have been installed before the modification was made, a further investigation of this problem was warranted.

On the basis of this possibility, a test was initiated both to operate all installed 1-inch 3000-pound globe valves through a complete travel using a torque wrench, and to record the torque readings. No notable deviations from the expected normal running torques were found in any of the valves. Therefore, it was concluded that this was an isolated case and that similar failure was not likely to occur in the near future.

The failed valve was replaced with a spare valve in stock. The stem, disc, and keys of the original valve have been replaced. The replacement valve was hydrostatically tested at 3750 psi with the disc seated at the prescribed torque.

In August, 1960 the 1B loop 3-way selector valve (15-H12-2) was isolated from the valve operation system due to internal leakage and was disassembled for inspection and repairs. This 1-1/2-inch selector valve is designed for water use, rated for 3000 psig at 220°F, and constructed of 304 stainless steel with stellite wearing surfaces. Inspection of the valve immediately after disassembly indicated that all its internal surfaces were free of foreign materials. The water inlet seal located in the upper portion of the valve was badly eroded over an area of approximately 150 degrees of its seating surface, indicating a large leakage area. A new seal and teflon back-up rings were installed. The seals for the water outlet sides of the valve, which direct water to the opening and closing ports of the pilot valve (15-H14-2), were found to be damaged and were replaced along with the teflon back-up rings. The top surface of the valve disc was found to be damaged and was replaced along with the teflon back-up rings. All internal surfaces of the valve were cleaned. The valve was reassembled and returned to service.

In September, 1960 the flash tank inlet 3-way selector valve (15-H12-8) was isolated from the valve operating system due to internal leakage and was disassembled for inspection and repair. This 1-inch selector valve is designed for water use, rated for 3000 psig at 220°F, and constructed of 304 stainless steel with stellite wearing surfaces.

All valve seals showed signs of wear and were replaced. The valve disc was severely worn and was replaced with a spare disc. The O-rings appeared to be in good condition, but were replaced as a preventative measure. The seal springs were badly deformed and all six were removed and replaced with new springs. The valve stem was in good condition and free from blemishes. Inspection of the valve immediately after disassembly indicated that all internal surfaces were free of foreign material. There was some very slight pitting present, but this pitting was scattered and therefore not considered serious. All internal surfaces of the valve were cleaned and the valve was reassembled and returned to service.

In November, 1960 the 1A loop flow d/p cell bypass valve (05-H16-24) was removed from service to be disassembled for inspection and repair. The 1-inch capped manual globe valve is rated for 3000 psig at 600°F and constructed of 304 stainless steel with stellite seating surfaces. It was suspected that the stem had broken and the disc had fallen on the seat, because the 1A loop flow d/p cell bellows would rupture each time the cell was hydrostatically tested. Disassembly and inspection of the valve disclosed that neither the seat nor the disc had any appreciable wear on its surface. However, the seat and disc did not have a smooth seating area.

Only about 30 percent of the surfaces were mating instead of the full surfaces for which the valve was designed. The valve was repaired by lapping-in the disc and seat with a special lapping tool, reassembled, and returned to stock. The original valve was replaced in the system with a spare valve. No further rupture of the cell bellows has occurred.

11. Turbine Trip Latch Problem

When the simulated weekly operational check of the overspeed trip was performed, the trip latch lever failed to reset with subsequent shutdown of the turbine on numerous occasions. In the performance of this weekly operational check, the auto-stop reset lever was held in the latched position to prevent the loss of the unit. The purpose of latching the auto-stop reset lever was to override the trip weight, which is thrown out from the shaft by centrifugal force, striking the overspeed trip trigger and causing the unit to actually trip due to overspeed of the turbine.

During a shutdown period in June, 1960 the overspeed trip mechanism was disassembled, inspected, and repaired. The overspeed trip valve reset and release cam was found to be worn and was replaced. Upon reassembly of the trip mechanism, it was also discovered that there was not sufficient clearance between the overspeed trip trigger and overspeed trip body. The overspeed trip trigger was removed and metal was removed from the trip trigger in the area that contacts the trip body.

The trip mechanism was then reassembled and the desired clearance obtained. Since then, the overspeed trip mechanism has worked satisfactorily when performing the simulated weekly operational check.

12. Differential Cell Calibration

A periodic calibration program for the 52 primary plant d/p cells was undertaken during Seed 2 life to determine the amount of drift that had taken place in the cells. The test involved the use of calibrating rigs to assure accuracies of 0.1 inch H_2O between the high and low legs. At times the isolation, or bypass, valves for the cells leaked through and liquid nitrogen freeze plugs had to be maintained on the lines to permit calibration. Very slight leakage of the bypass valve is tolerated during operation when there is sufficient flow of water to maintain the head across the cell, but during static calibration no leakage at all can be tolerated. In the case of the d/p cells used on core instrumentation, no isolation valves are provided; therefore, freeze plugs had to be maintained at all times during calibration. Maintenance of the freeze plugs required an extra man on the calibrating teams. Venting the cells was sometimes difficult because of gas being liberated from the primary water. During Seed 2 operation, the internal relief valves in the d/p cells sometimes failed to reseat. The cells would indicate reasonably well in operation, but were impossible to calibrate because a static head could not be maintained across the cell. The relief valve required removal of the cell from the system, cutting the seal welds on both the top and bottom caps of the cell body, and rebuilding the cell. Then the cell was welded shut, with calibration before and after welding and rewelded into the primary system.

13. Reactor Plant Instrumentation

This section includes all instruments associated with the reactor and reactor plant auxiliary fluid systems. Due to failure of many of the indicator-receivers during Seed 1 life as a result of inadequate lubrication and heat, all 57 units were torn down, cleaned, and rebuilt during Seed 1 - Seed 2 refueling. During Seed 2 lifetime 12 units failed because of parts freezing due to inadequate lubrication and two units failed when the magamp power supplies failed. This type of power supply failure had not been experienced during Seed 1 operation. One final problem is associated with the oscillation dampers which are provided on flow instruments to prevent them from oscillating in one spot and causing excessive wear on parts. These units frequently failed during Seed 1 life and into Seed 2 operation. The trouble was remedied by using higher rated diodes in the bridge rectifier circuit. These diodes had not been commercially available until that time.

During Seed 1 - Seed 2 refueling, the temperature monitors were replaced with monitors of a newer design, mainly due to the high cost of repairs to the stepping switches. New stepping switches for the new units were obtained at approximately 10 percent of the cost of reconditioning the switches for the old monitors and lasted 6 to 9 months; however, the switches had to be soldered in (180 connections) instead of plugged in as in the original monitors. The monitor design was modified to slow down the scanning time to one point every 1-1/2 seconds, thus enabling the switch to last 36 to 54 months since switch life is directly proportional to operations.

The valve position indicating system uses E-core differential transformers to detect sealed hydraulic valve movements. The output is fed to magamp receivers which amplify and interpret the signal to indicate whether the valve is open or closed. During Seed 1 - Seed 2 refueling the detectors were reconditioned to correct for damaged insulation and contact corrosion caused by high ambient temperature (130-150°F) in the vicinity of the detector and by heat conduction from the valve. During Seed 2 operation, the detectors gave no trouble, but the magamp receivers drifted and, in one case, caused a safety shutdown. Better ventilation has been installed in the receiver racks, and the receivers are being modified to permit checking for receiver drift during operation.

14. <u>Radioactive Waste Disposal System Level Instrumentation</u>

In the waste disposal plant there are 27 strain gage type d/p cells feeding into one read-out instrument of the capacitive balance type installed in the waste disposal building. The seasonal rebalancing of cells is still required as during Seed 1 operation but, in addition, it has been noted that temperature changes cause zero drift of the cells. The 21 cells installed in instrument cabinets outside the building have a high rate of failure, with repairs costing approximately 70 percent of the price of a new cell. Of the 21 outside cells, 8 had to be returned to the manufacturer for repairs during Seed 2 operation. Because of troubles, the strain gage type d/p cells will be replaced by E-core differential transformer type cells for tank level indication and by level switches for sump levels.

15. <u>Operational and Safety Radiation Monitoring Systems</u>

The operational radiation monitoring system monitors various critical areas within the reactor plant and has read-out instruments in the main control room. Channels 9 through 12, air particle detectors, were modified at the beginning of Seed 2 operation. The filter paper drive and filter paper were changed. This modification resulted in a substantial reduction in equipment failures.

Equipment, Personnel, and Procedure Improvements

Tool and equipment cleaning for transfer from a controlled or contaminated area to a clean area was time consuming in the maintenance program. Initially, tools were categorized either as clean tools or as contaminated tools and kept in separate areas of the reactor plant. This arrangement worked satisfactorily for a short period of time, but inevitably an interchanging of tools was encountered. To overcome this difficulty, the two tool storage and issuing areas were combined into one; thus, all reactor plant tools were treated as contaminated.

Transfer of a tool or piece of equipment from the reactor plant required that it be cleaned until the contamination level was less than 100 dpm/ft^2 beta gamma regardless of destination or use. In some cases, the item was merely to be inspected or dismantled and then returned to a contaminated area. Removal of the equipment to the decontamination room and assignment of men to clean the item, if routine decontamination was not being performed at the time, was often necessary. Many times a job was delayed for several days while some tool or piece of equipment was being decontaminated. The subsequent purchase of an ultrasonic decontaminating bath considerably reduced the time necessary for tool and component decontamination during Seed 2 operation. Fast and effective decontamination of even the most intricate parts was achieved with relative ease in a short time. A further improvement was made by the installation of a second contaminated tool room in the reactor plant container. This facility reduced the manhours necessary to obtain the proper tools when work was being performed in the plant containers. These tools are periodically checked for high contamination levels and are decontaminated only when required. Previously, the men had to obtain the tools from the tool room located in the fuel handling building and return them to the tool room after they were decontaminated.

The purchase and installation of additional machinery, such as a horizontal boring mill (replaced outdated boring mill), punch and shear, shaper, and milling machine, enabled more machine work to be done at the site rather than at a distant machine shop. Also, certain items that were difficult to decontaminate were machined under health physics supervision in the contaminated machinery area.

Previous work on equipment at a conventional power station did not normally involve the use of procedures, because experienced personnel who had maintained the equipment over a period of years were able to perform the work unaided and to train newer men. Since the maintenance men were unfamiliar with much of the reactor plant equipment, use of detailed procedures became mandatory for disassembly, repair, or assembly of equipment. At first this slowed down their pace as did the donning of the correct anti-contamination clothing. During Seed 2 operation, however, the men have become more proficient in the use of detailed procedures to perform maintenance work in the reactor plant section and have made use of experience gained in Seed 1 operation. Maintenance time was reduced on individual instruments since the instrument personnel had become familiar with use of test equipment and with the normal troubles to be expected from equipment. They were also more familiar with the individual instruments and their circuitry; therefore, troubleshooting was much easier.

The turnover of maintenance personnel has been practically nil for this station during Seed 2 operation. The addition of two maintenance foremen during the latter half of Seed 1 operation has improved the working efficiency of all personnel. With three foremen now assigned to maintenance, one foreman handles the mechanical maintenance in the turbine-generator plant, and the other mechanical maintenance foreman handles work in the reactor plant. Their maintenance men are subject to change in working areas, dependent upon the work load, if an emergency arises in the reactor plant or turbine-generator plant. The third foreman is assigned to handle all electrical maintenance problems in both plants.

Experienced personnel turnover in the instrument group was limited to one mechanical instrument man. Thus, all personnel were trained and familiar with the plant and its instruments and could work at top efficiency.

During Seed 1 operation considerable time was lost when spare material and equipment were not available. This condition was due to the unexpected frequency of repairs to various equipment. Also, various material substitutes were tried to see if they would have a longer life. From the experience gained during Seed 1 operation, adequate spare material and equipment were kept in the storeroom stock for Seed 2 operation. The minimum stock level specified for various items was based on the actual frequency of repairs for various plant equipment. As a result, the loss of time was reduced on various maintenance jobs because of the availability of material.

Additional equipment for handling material was procured during Seed 2 life. An additional mobile crane was purchased to facilitate the handling of heavy equipment. Two additional fork lift trucks were purchased and greatly reduced the time lost in handling bulky material on certain maintenance jobs.

During the latter portion of Seed 2 operation, a central shipping and receiving station was established. This station reduced the loss time on maintenance jobs, especially when equipment was removed and sent out for modification or when new equipment was received for installation. Previously, the equipment was usually prepared for shipment at the point of removal and men were moved from other jobs to load the equipment if it was not shipped immediately.

Since the plant operated at full power during most of Seed 1 - Seed 2 life, with only a minimum number of shutdowns of short durations for testing, the repair or overhauling of nonessential items within the plant had to be postponed. Maintenance of this equipment will therefore be performed during the forthcoming refueling operation.

PART II

REACTOR PHYSICS PERFORMANCE

CHAPTER 1

COMPARISON OF SEED 1 AND SEED 2 NUCLEAR BEHAVIOR

Introduction

The nuclear behavior of PWR Core I Seed 2, as determined in periodic physics test results performed through Seed 2 lifetime, has provided information confirming behavior observed during Seed 1 operation. Significant features observed during Seed 2 lifetime included a pronounced increase in the blanket reactivity. Higher blanket power fraction, caused by the enhanced reactivity of the natural uranium dioxide blanket with continued irradiation, resulted in a substantially longer Seed 2 lifetime than expected. Furthermore, spontaneous xenon oscillations during full power operation of Seed 2 were not observed, in contrast to Seed 1 experience. Unusual phenomena occurring during Seed 2 operation were the variation of the full power temperature coefficient with core age and the transient rod motion associated with initial withdrawal of control rod groups while at power.

The physical characteristics which distinguished the nuclear behavior of PWR Core I Seed 2 from PWR Core I Seed 1 included (1) an increased seed fuel loading, (2) use of two fuel zones in each Seed 2 subassembly to decrease local power peaking, (3) inclusion in one of the fuel zones of small amounts of distributed boron to act as a burnable poison, and (4) differences in water-to-nonwater volume ratio due to slightly different mechanical designs. The objective of the higher Seed 2 fuel loading was to obtain a longer seed operating lifetime; the natural boron was included to help control the reactivity of the core during the first several thousand hours of operation and to help suppress power peaking. All other components of the core remained the same for Seed 2 as in Seed 1, except for the replacement of several blanket fuel assemblies and a control rod, which were removed for destructive analysis during refueling operations. In order to acquire operating experience with lesser artificial neutron source strength during start-up, only two polonium-beryllium neutron sources were installed for Seed 2 operation as contrasted with four neutron sources used with Seed 1.

The reactivity contribution of the irradiated natural uranium dioxide blanket material was different from that during Seed 1 by virtue of the formation of plutonium and fission products in the blanket during Seed 1 operation. At the conclusion of Seed 2 operation, the blanket fuel region is calculated to have experienced an average burnup of approximately 4500 megawatt-days per ton of natural uranium dioxide (MWD/T) and a peak burnup of about 20,000 MWD/T, whereas, following Seed 1 operation, the comparable values are approximately 2000 MWD/T average and about 9000 MWD/T peak burnup. Operation of the core with Seed 2 installed has indicated the important contribution of the blanket reactivity lifetime characteristics to the over-all behavior of the seed-blanket core, as indicated below.

The reactivity lifetime of PWR Core I with successive seeds installed depends on the relative depletion of the seed and irradiated blanket. Thus, the power sharing between the enriched seed and natural uranium blanket of PWR Core I is a parameter of considerable significance. Although there was considerable scatter in the data, the blanket power fraction, as inferred from exit water thermocouple data, appeared slightly higher during Seed 2 operation than predicted on the basis of one and three-dimensional fuel depletion calculations.

This power sharing behavior is consistent with the substantially longer Seed 2 lifetime than predicted, which tends to suggest a reactivity behavior, with extended irradiation, of the natural uranium blanket fuel that is more favorable than calculations indicated. The full power equilibrium xenon lifetime of Seed 2 was anticipated to be longer than for Seed 1 because of the increased seed fuel loading over Seed 1, but the 7900 equivalent full power hours lifetime actually attained for Seed 2 exceeded expectations based on fuel depletion calculations by about 1000 equivalent full power hours. In contrast, the same type of calculations gave a slight overestimate of the Seed 1 lifetime.

It is of interest to compare physics test results from Seeds 1 and 2 in order to understand and evaluate the differences and similarities observed. Since the seed region primarily determines the reactivity characteristics of a seed-blanket reactor, most of the Seed 1 and Seed 2 physics test results can be expected to be similar because the physical differences between the seeds have a small over-all effect on core reactivity. The effect of increasing the seed loading from 75 to 90 kilograms is nearly compensated for by the inclusion of 170 grams of natural boron. Although the blanket power sharing and reactivity lifetime are the quantities principally affected by the blanket behavior, the accompanying influences from seed differences preclude complete separation of seed and blanket effects. However, the comparisons presented below do permit some qualitative observations to be made.

Comparison of the periodic zero power physics test results for Seed 1 and Seed 2 has not shown any substantial differences. The trend towards less negative temperature coefficients at zero power with increasing core age for Seed 2 is similar to the behavior observed during Seed 1 operation. Due to the increased fuel loading of Seed 2, the average seed thermal neutron flux levels as inferred from the xenon transient tests have been generally lower, as expected, than comparable Seed 1 flux levels. Measured Seed 2 control rod reactivity worths were similar to those measured on Seed 1, so that no pronounced loss in rod worth is evident from the depletion of the neutron absorbing isotopes of the hafnium control rods.

A phenomenon not observed during Seed 1 life was the variation of the full power temperature coefficient with core age. Seed 2 did not exhibit any tendency toward spontaneous xenon oscillations during power operation as in Seed 1, Reference (2).

An unusual phenomenon occurring in Seed 2, and not in Seed 1, was the oscillatory rod motion associated with the initial withdrawal of the Group III and Group IV control rods while at power. These topics are discussed more fully in later sections.

Extensive programs are in progress to evaluate further the reactivity behavior and depletion effects of the irradiated blanket, depleted seed, and hafnium control rods. Following Seed 2 operation, several blanket fuel assemblies were removed from the core for evaluation. Selected blanket bundles will be placed successively in a test reactor to determine the relative reactivity behavior of these bundles for comparison with calculated bundle-wise reactivity variations in the PWR Core I blanket. Certain blanket fuel rods from these bundles, selected depleted seed fuel plates, and control rod samples will be destructively analyzed to determine isotopic compositions, which will then be compared with calculated values. A similar destructive analysis program was conducted on blanket fuel rods removed following Seed 1 operation (Reference 1). Thus, an extensive physics program, including experimentation with the Shippingport reactor and subsequent destructive analysis of some of its components, is being carried out for PWR Core I Seed 2 in order to promote understanding of the lifetime nuclear characteristics of a seed-blanket reactor with successive seeds installed.

CHAPTER 2

PHYSICS TEST PROGRAM

Introduction

With the completion of the first refueling, the physics testing program initiated for Core I Seed 1 was continued for the lifetime of Seed 2. The objective of this program was to obtain measures of the reactivity and power distribution characteristics of the reactor as a function of core lifetime associated with extended power operation. The results were used to provide indications of the seed lifetime to be realized and to predict future core performance. Experimental information derived from physics testing also can be compared with analytical results to determine the validity of calculational design models. In areas where significant differences between the results obtained by experiment and by calculation are observed, modifications to the calculational technique can be attempted to improve the design model.

Physics Testing Conditions

Associated with reactor physics testing are several standard plant conditions for full power, reduced power, and zero power tests. Full (100 percent) power operation for Seed 2 was defined as an output of 231 core thermal megawatts at an average coolant temperature of 500°F, with the primary coolant pressure maintained at 1800 psi. For reduced power operation at the end of seed life, the average coolant temperature was reduced to 475°F, and extended runs were made at power levels of 75 and 50 percent of full power capability. The "zero power" condition for physics testing is at specified temperatures and at power levels of up to 50 thermal kilowatts.

Reactor Instrumentation and Control

Precise reactor instrumentation and control of core and plant conditions during physics testing is necessary for the performance of the test and acquisition of accurate experimental measurements. The circumstances under which the physics tests are performed involve numerous interactions between the reactor proper and the plant. Two predominant effects, temperature and pressure variations, which directly influence nuclear behavior are accounted for through the use of temperature and pressure corrections.

Due to the large temperature coefficient of reactivity of PWR, the variation of coolant (moderator) temperature must be measured to at least ± 0.1°F to insure precise reactivity measurements. For this purpose platinum resistance thermometers are installed in each of the coolant loops. Readings from these resistance thermometers are displayed on precision self-balancing Mueller bridge recorders. With these instruments, absolute core inlet temperature indication within ± 0.1°F is obtained, and differential changes of ± 0.01°F are observable. Maintaining constant reactor coolant temperature for certain physics tests is difficult because the primary coolant pumps are operable at only two speeds to provide heat input. However, average core temperature is maintained to within approximately ± 0.5°F during physics testing.

Additional detectors located in the pressurizer and in each coolant loop measure the primary coolant pressure. The smallest scale division on the narrow range indicator is 10 psi. In practice, pressure control is maintained at ± 20 psi for zero power testing and ± 30 psi for power operations. The location of each of the four primary coolant loops is indicated in Figure II-1.

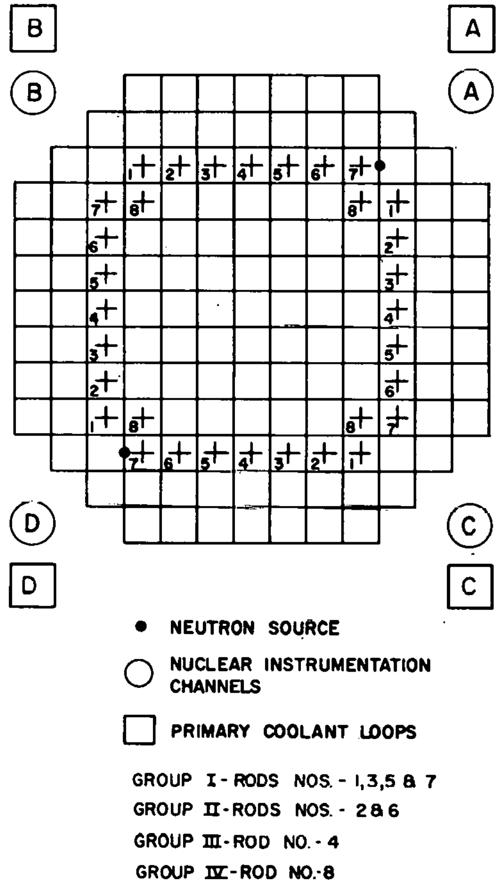

Figure II-1. PWR Rod Groups and Instrumentation.

Special fission counters which operate independently of the plant nuclear instruments are inserted into two of the detector wells in the neutron shield tank for zero power physics testing. The placement of the fission counters is such that the reactor period measurements at minimum power level are not influenced by the installed neutron sources and the reactor heating of the moderator. These special fission counters are used since normal nuclear instrumentation may provide less precise reactor period data. Output signals from the fission counters are fed into non-overload amplifiers connected to high-speed scalers. The interval during which counts from the scalers are accumulated is selected and controlled with another preset digital counter.

In order to prevent damage and induced radioactivity during power operation, the special fission counters must be inserted before and removed after each series of zero power physics tests. However, the procedure for the xenon transient test does not allow sufficient time after shutdown to insert the special fission counters, and the regular nuclear instrumentation is then used to obtain reactor period measurements. During xenon transients, period measurements are made by amplifying the compensated ion chamber current from one of the instrument channels and measuring the time interval required for an e-fold change in the power level as displayed on a chart recorder. Figure II-1 shows the location of the nuclear instrumentation channels which monitor symmetrical quandrants of the core.

The accumulated number of equivalent full power hours (EFPH) of operation is read from a recorder which displays the integral of the indicated percent core power level as a function of time. The indicated power is taken from the output of one of the linear amplifiers associated with the nuclear instrumentation channels. All instrumentation channels are calibrated periodically so that 100 percent power corresponds to 231 core thermal megawatts during normal operation at 500°F average coolant temperature and 1800 psia. Because of this calibration, the recorded number of EFPH is estimated to be uncertain by no more than ± 3 percent due to instrument drifts between calibrations and due to changes in the neutron flux level as registered by the neutron detectors.

Core I contains 32 hafnium control rods which are divided into eight symmetrical subgroups of four rods each. These eight subgroups of rods are arranged into four normal rod programming groups which are withdrawn in sequence as required for criticality control. The identification by number of the individual control rods and the rod groups is shown in Figure II-1. The normal sequence of controlling rod group withdrawal during power operation is listed in Table II-A.

Each control rod is moved by means of a lead screw attached to a variable reluctance motor powered by an inverter which converts 235 volt direct current into variable frequency 3-phase alternating current. Approximate control rod positions are obtained from the normal rod position indicators and differential rod motions are measured by observing the rotation of the inverter faceplates. Scales attached to each of the eight normal and two spare inverter faceplates calibrated in 0.01 inch divisions provide an accurate means of positioning the control rods.

During normal control rod operation, the rods can be withdrawn to an upper programmed limit of 69 inches above the bottom of the core. With manual control, the rods can be withdrawn to approximately the full core height of 72 inches. The terminology fully inserted indicates that the rods are at or near the bottom of the core; fully withdrawn indicates that the rods are at the upper program limit (approximately 69 inches), unless otherwise indicated.

Prior to the performance of zero power physics tests, the control rods are calibrated by inserting the control rods in each subgroup to their lower limit and re-zeroing the individual rod position indicators. This procedure locates the position of all control rod tips to within ± 0.25 inch of each other. Since the inverter speeds within a rod group vary from subgroup to subgroup, the relative alignment of the control rods may change during the performance of a test. The average rod height is defined as the average of the individual rod position indicator readings of the controlling rods.

Various control rod patterns can be established by means of two spare inverters, to which individual rods or a subgroup may be transferred for other than normal rod programming configurations. This added flexibility of the rod control system has been used during Seed 2 in the performance of special tests.

TABLE II-A

NORMAL SEQUENCE OF ROD WITHDRAWAL

Rod Group	Total Rods Per Group	Subgroups*
I	16	1, 3, 5, 7
II	8	2, 6
III	4	4
IV	4	8

* Each subgroup is controlled by a single inverter.

Tests Conducted

The physics testing program provides for periodic and continuous performance of tests at zero and operating power to determine critical control rod positions and reactivity worths, excess reactivity, temperature coefficients of reactivity, xenon transient effects, and reactivity lifetime behavior. Listed in Table II-B are test specification numbers, performance dates, type of tests, and purpose of each test as performed throughout Seed 2 lifetime. The combined results of these physics tests provide experimental information to permit analysis and evaluation of over-all core nuclear behavior throughout lifetime.

Prior to initial power operations, a series of zero power physics tests was conducted on Seed 2 to examine the initial nuclear characteristics of the core. Quantities measured in the start-up tests were the temperature defect in reactivity between ambient and operating conditions, reactivity symmetry of the core, excess reactivity both hot and cold, and the clusterwise variation of the one-rod-withdrawn shutdown reactivity at ambient temperature.

Following the initial reactivity testing, the core was operated continuously throughout seed life except for periodic shutdowns for zero power testing. The periodic zero power group of tests included measurements of critical control rod positions and reactivity worths at ambient and operating conditions, temperature coefficients of reactivity, and xenon transient effects. In addition to the periodic zero power tests, the rod withdrawal rate, temperature effects on core performance, reactivity lifetime, and seed-blanket power distributions were obtained during periods of power operation. Prior to the end of Seed 2 life, a special axial flux perturbation test was conducted to determine if an axial power oscillation could be induced in the core.

At the end of full power capability, a series of zero power tests were performed to examine the reactivity symmetry and the remaining excess reactivity, and to determine fuel depletion and fission product poisoning effects.

By reducing the moderator temperature and operating at reduced power levels, the energy utilization of the core was extended beyond the equilibrium xenon and samarium condition full power operating capability. During this period of reduced power operation, an intentional xenon radial oscillation tes was performed to observe possible power distribution fluctuations. At the conclusion of a 75 percen

TABLE II-B

SEED 2 PHYSICS TESTS

DLCS Number	Title	EFPH on Seed 2 Prior to Test Performance	Power Level During Test	Purpose
14901	Control Rod Position for Criticality	0, 3560, 6145, 7528	Zero	To determine critical bank heights and bank worths for various rod configurations at ambient temperature.
14902	Control Rod Position for Criticality	0, 3560 6145, 7528	Zero	To determine critical bank heights and bank worths for various rod configurations at normal plant operating temperature and pressure.
15001	Calibration and Intercomparison of Control Rods	0	Zero	To determine reactivity symmetry of the core and rod worths.
15003	Calibration and Intercomparison of Control Rods	7528	Zero	To determine the relative fuel depletion of symmetrical sections of the core.
15101	Determination of Coefficients of Reactivity	0, 3560 6145, 7528	Zero	To determine the temperature and pressure coefficients of reactivity.
15102	Determination of Coefficient of Reactivity	150, 750 1117, 1688 2210, 3065 3515, 4810 6115, 7415	Operational	To determine the temperature coefficient of reactivity in the power range.
15601	Xenon Transient Test	150, 750 1560, 2240 3560, 4830 6145, 7528	Zero	To determine the reactivity of the reactor during transient xenon conditions throughout core life.
15604	Samarium Transient Test.	7762	Zero	To determine the reactivity effect of peak samarium conditions.
16001	Initial Approach to Criticality	0	Zero	To bring the cold, clean reactor core safely to criticality.

TABLE II-B (Cont'd)

SEED 2 PHYSICS TESTS

DLCS Number	Title	EFPH on Seed 2 Prior to Test Performance	Power Level During Test	Purpose
36201	Reactivity Lifetime	Continuous	Operational	To obtain instrument readings from which total core power output, core power distribution, core stability, reactivity lifetime, and gross electrical output may be determined.
38801	Axial Flux Perturbation Test	7160	Operational	To determine the reactivity fluctuation resulting from a non-equilibrium axial xenon distribution induced by altering the control rod configuration while at power.
38802	Radial Flux Perturbation Test	7600	Operational	To determine the variation of plant parameters when a radial power tilt is introduced across the core and to observe any power oscillation induced by a varying radial xenon concentration.

power run, the reactivity variation due to transient xenon and samarium poison was observed for about 240 hours.

Analytical Methods Used in Data Evaluation

All reactivity values determined from the various tests were inferred from measurements of positive reactor periods. The reactivity values inferred from period measurements are calculated using a fractional delayed neutron yield of $\beta^{25} = 0.0064$, and appropriate delayed emitter yield and decay constants. The $\overline{\beta}/\beta^{25}$ correction factor for the relative importance of delayed and prompt neutrons was evaluated using standard calculational techniques in a three-dimensional diffusion theory calculation in four energy groups. As discussed in Reference 2, the result of this calculation indicated a small variation in the value of $\overline{\beta}/\beta^{25}$ over core lifetime. However, no β/B^{25} correction was applied to the experimental data as the maximum value of the β/β^{25} correction is of the same order of magnitude as the experimental uncertainty of the measurements.

The inferred reactivity values obtained from measurements of reactor period are corrected for slight variations in reactor temperature and pressure by appropriate correction factors. These correction factors are used to adjust the inferred reactivity values to uniform reactor operating conditions of temperature and pressure. The experimental values of reactivity as corrected for small variations in temperature and pressure are correlated by the method of least squares in order to minimize the effect of scatter in the data.

Xenon and Samarium Corrections

Measurements reported herein have not been adjusted for the effects of xenon and samarium poisoning variation which occurred during the tests since the magnitudes of such corrections are dependent upon the calculational model used in the evaluation. Measurements other than xenon transient tests, in which xenon and samarium would have an influence, are performed after a sufficient interval of time has elapsed since the previous power operation, such that the Xenon 135 concentration is negligible and the concentration of stable Samarium 149 has reached a maximum.

Nuclear Design Calculations

Calculations were performed to predict the nuclear behavior of Seed 2 using several physical models with the CANDLE, TURBO, and DRACO criticality and fuel depletion codes, References 3 and 4. Reference 5 contains a description of the PWR three-dimensional calculations which have been performed using the DRACO code, and Reference 6 and 7 contain description of the two-dimensional TURBO code and the CANDLE code calculations.

Comparisons are made, where applicable, in following sections between the calculated results using the above codes and the experimental results in an effort to assess the validity of the physical models employed. Some of the factors which influence the calculated results are: number of dimensions, number of energy groups, size of mesh divisions, diffusion theory parameters representing homogenized regions, and the value used for the fission product absorption cross-section per fission. As new theoretical developments occur, they are incorporated into nuclear design calculations, so that revisions in calculated results occur frequently. A comprehensive report of the calculations applied and results obtained for Seed 1 is contained in Reference 8.

CHAPTER 3

PRESENTATION AND INTERPRETATION OF PERIODIC TEST RESULTS

Startup Testing

Initial Approach to Criticality

The initial approach to criticality of Core I Seed 2 was performed on April 6, 1960, and criticality was achieved with the number 1 and 7 control rod subgroups positioned at 24.7 inches. The reactor moderator temperature was 140°F, and primary pressure was maintained at approximately 380 psig. During the approach to criticality, the control rod bank was withdrawn in one-inch increments until significant neutron multiplication was observed on the source range nuclear instruments. The rod withdrawal increment was reduced as the observed count rates increased due to increased neutron multiplication. Inverse count rates of the four instrument channels were plotted as a function of rod position to predict by extrapolation a critical height prior to the next incremental rod withdrawal.

Individual control rod locations, nuclear instrumentation channels, and installed neutron source locations are shown in Figure II-1. The presence of only two neutron sources created no instrumentation or monitoring problems during the initial approach to criticality.

The neutron multiplication observed while approaching criticality during these initial reactivity measurements was adequately indicated by the nuclear instruments, although the two detectors farthest from the neutron sources gave count rates approximately one-tenth that from detectors near a source.

One-Rod-Removed Reactivity Measurements

Measurements of the one-rod-removed reactivity were made for each individual seed cluster location. Except for the number 8 inset corner locations, the measurements were made for each cluster with its control rod fully withdrawn and the adjoining rod in the clockwise direction withdrawn to the critical height. The integral reactivity worth of the adjoining control rod from the fully inserted position to the measured critical height was interpreted as the one-rod-removed reactivity of the cluster.

For the inset corner cluster measurements, the adjoining rod located clockwise was positioned at nine inches and the inset corner rod withdrawn to the critical height. The integral worth of the adjoining rod from zero to nine inches, minus the excess reactivity remaining in the inset rod, was interpreted to be the one-rod-removed reactivity of the inset cluster. Integral reactivities of the control rods were inferred from a reactivity worth curve determined in a Seed 2 mock-up at the Bettis PWR Critical Facility. The results measured at 135°F for the one-rod-removed reactivity for each cluster are given in Table II-C. Although a systematic reactivity variation is apparent among the numbered cluster locations in each core quadrant due to the inherent reactivity properties of the core geometry, no appreciable reactivity variation is evident between clusters in corresponding positions in each of the four quadrants.

Reactivity Symmetry Measurements

Reactivity symmetry measurements of the corner regions of the seed were made by measuring the critical rod bank height and reactivity worth with the inset corner rod and the two adjoining control rods banked. The relative reactivities of each of the four corner seed regions were determined by adjusting each measurement with reference to the southeast quadrant. Each difference in rod height times the measured reactivity worth of the corresponding quadrant yielded a relative reactivity comparison of each core quadrant. Shown in Table II-D are results of core quadrant relative reactivity symmetry measurements at zero EFPH.

Full core two-dimensional (radial) calculations were performed which represented the variations in water-to-non-water volume, fuel concentration, and boron concentration for each individual Seed 2 cluster as manufactured and selectively placed in the core. These calculations indicated a power asymmetry across the core of less than one percent. This result is compatible with the above measurements, which indicate no appreciable reactivity asymmetry.

TABLE II-C

SEED 2 INITIAL ONE-ROD-REMOVED REACTIVITY AT 135°F

Cluster Number	SW Quadrant (% $\Delta\rho$)	SE Quadrant (% $\Delta\rho$)	NE Quadrant (% $\Delta\rho$)	NW Quadrant (% $\Delta\rho$)
1	-0.68 *	-0.80 *	-0.74 *	-0.87 *
2	-0.74	-0.80	-0.83	-0.83
3	-1.32	-1.32	-1.35	-1.29
4	-1.35	-1.45	-1.27	-1.51
5	-1.16	-1.11	-1.16	-1.11
6	-0.80	-0.98	-0.53	-0.78
7	-0.71	-0.65	-0.51	-0.68
8	-0.20	-0.00	-0.00	-0.03

* The uncertainty associated with each value is estimated to be less than ±5% of the individual value.

TABLE II-D

SEED 2 INITIAL COMPARISON OF RELATIVE QUADRANT REACTIVITIES

Quadrant	Critical Height of Rods 1, 7, 8 (in.)	Rod Bank Worth ($\Delta\rho / \Delta h$) (10^{-4}/in.)	Relative Reactivity (% $\Delta\rho$)
SW	16.0 *	70.6	-0.21
SE	15.7	58.1	0
NE	15.5	56.1	0.11
NW	15.6	62.8	0.06

* The uncertainty of absolute rod heights is estimated to be $\pm$0.25 inch.

Pressure Coefficient of Reactivity Measurement

The pressure coefficient of reactivity as measured at the beginning of life between 1220 and 1170 psig was $1.80 \pm 0.04 \times 10^{-6}$ $\Delta\rho$/psi. Using a constant rod bank height, changes in reactivity were determined from reactor period measurements as the moderator pressure was incrementally increased over the range of measurements.

Additional results from tests performed not only at the beginning of life but also throughout Seed 2 life are presented in the following section.

Lifetime Testing

Critical Rod Position and Reactivity Worth Measurements

At various times during core life, measurements were made of critical rod positions and reactivity worths for various control rod configurations. The results for these measurements are summarized in Table II-E for ambient and 500°F temperature conditions. Individual rod locations and groupings for the control rod configurations listed in Table II-E are illustrated in Figure II-2. Fewer measurements of critical bank heights and reactivity worths were made on Seed 2 than on Seed 1 because the lifetime variation in reactivity was expected to be similar to that found in Seed 1.

The values for the critical rod bank height and reactivity worth are obtained by the method of least square fitting of the pressure and temperature corrected experimental data. These measurements of rod worth are required for the determination of the excess reactivity of the core at ambient and operating temperature. No reactivity adjustment for samarium effects has been applied to the rod position and reactivity worth measurements. All of these measurements were made at or near peak samarium conditions by virtue of a time interval between shutdown and the measurements of at least 100 hours.

The reactivity worth measurements made throughout Seed 2 lifetime are plotted as a function of critical rod bank height in Figures II-3 and II-4 for the ambient temperature test and 500°F test, respectively. For the ambient test, measurements were made at temperatures between 130°F and 190°F. In each figure, a separate curve is drawn for each set of measurements throughout seed life. The differences between curves at successive measurements for 500°F and ambient temperature are

TABLE II-E

SEED 2 CONTROL ROD CRITICAL POSITIONS AND REACTIVITY WORTHS XENON-FREE AND APPROXIMATELY PEAK SAMARIUM CONDITIONS*

		Ambient Condition			Hot Condition, 500°F	
Rods in Bank	EFPH	Temperature (°F)	Critical Height (in.)	Rod Bank Worth (10^{-4}/in.)	Critical Height (in.)	Rod Bank Worth (10^{-4}/in.)
All rods	0	135	8.99 **	136.6 ± 3.2	12.17 **	106.0 ± 3.9
	3560	140	13.48	95.2 ± 6.1	18.12	55.6 ± 3.5
	6145	190	18.70	43.6 ± 0.9	26.49	26.9 ± 1.1
	7528	130	28.22	13.3 ± 0.3	47.71	8.3 ± 0.2
Groups I, II	0	135	11.33	91.6 ± 1.8	16.41	67.5 ± 1.3
	3560	140	20.51	39.5 ± 1.0	31.54	25.5 ± 2.7
Rods #3, 4, 5	0	135	16.09	55.2 ± 1.1	25.99	28.6 ± 0.6
	3560	140	32.89	17.6 ± 0.4	57.57	10.7 ± 0.2
Group I	0				29.61	19.9 ± 1.3
Groups I, II, III	6145	190	27.47	18.3 ± 0.6	44.14	12.7 ± 0.4
	7528	130	60.13	18.2 ± 0.6	Reactor	Subcritical
Groups I, II, IV	6145	190	22.62	28.0 ± 0.6	36.29	13.8 ± 0.3
	7528	130	49.43	6.6 ± 0.1	64.87	19.4 ± 0.4
Normal rod program						
Group II	3560				21.15	13.7 ± 0.4
Group II	6145				63.60	7.1 ± 0.3
Group IV	7528				16.95	3.4 ± 0.1

* Except for measurements at 0 EFPH.

** The uncertainty of absolute rod heights is estimated to be ±0.25 inch.

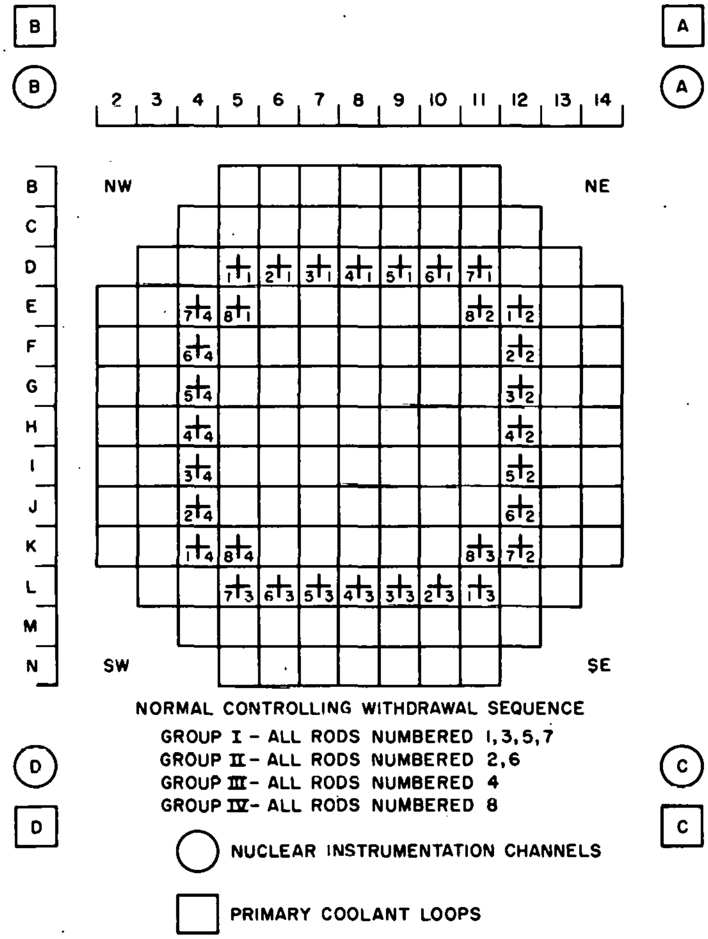

Figure II-2. PWR Core I Rod Groups and Instrumentation-Control Rod Numbers and Cluster Coordinates.

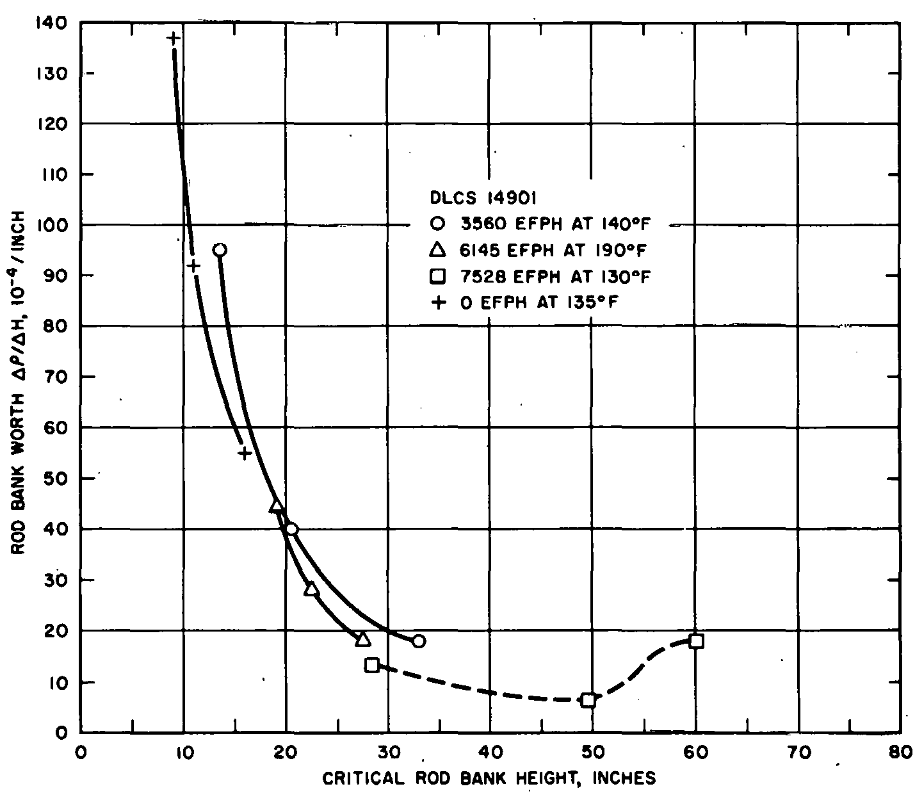

Figure II-3. Seed 2 Rod Bank Worth vs Critical Height at Ambient Temperature.

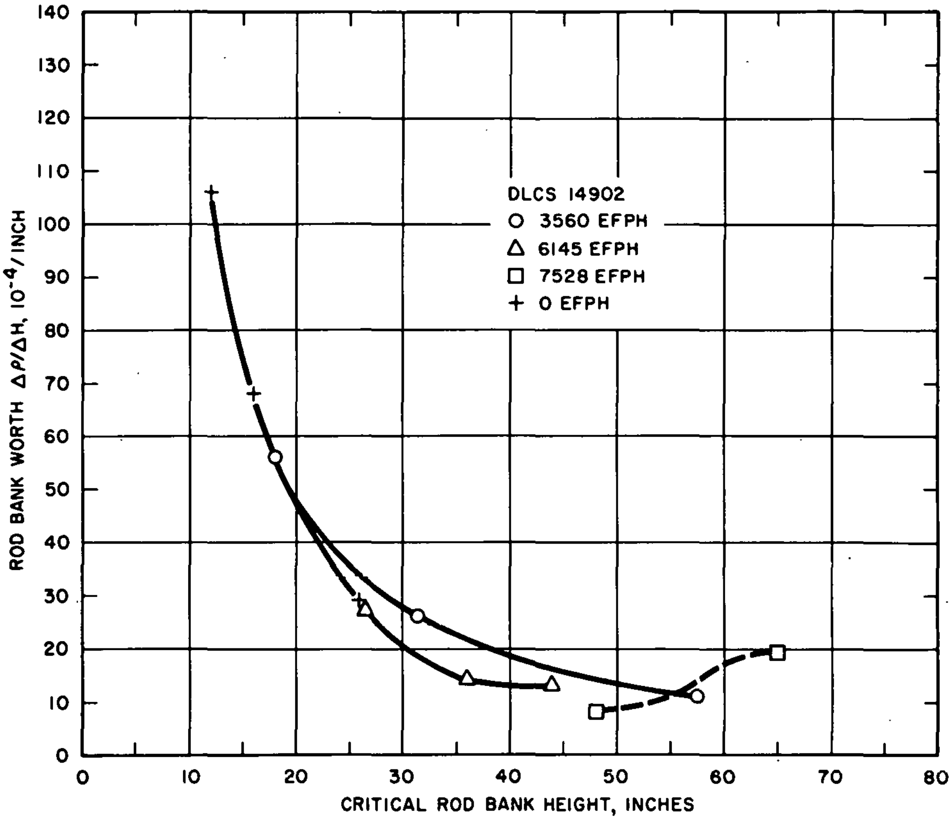

Figure II-4. Seed 2 Rod Bank Worth vs Critical Height at 500°F.

apparently the effects of changing fuel compositions between successive measurements. The increased rod bank worths measured at rod heights above 40 inches are a consequence of axial neutron flux peaking in the top of the core late in seed life, due to the relatively small depletion there, compared with that in the lower portion of the seed.

Xenon-Free, Maximum Samarium Excess Reactivity

A measure of the excess reactivity of PWR can be obtained by integrating to the full core height the rod reactivity worth measurements made at various control rod bank heights. As discussed in Reference 2, the reactivity worth is a unique function of the inverse cube of the core height and is independent of the effective multiplication factor of the core if certain parameters describing the neutron behavior (migration area, infinite multiplication factor, and reflector savings) are not spatially varying. Thus it is possible to plot the reciprocal cube root of the rod worth as a linear function of the core height. This linear relationship is illustrated in a typical plot of measurements made on Core I Seed 2, Figure II-5. By analytical integration of this relationship, the excess reactivity corresponding to a particular condition of temperature and EFPH may be calculated.

A correction to this calculated excess reactivity must be applied since partial moderator height measurements, as are usually performed with critical mock-up assemblies to determine an effective core height, are not possible in Core I. In place of water height measurements, the variation of the height of a bank of control rods must be used. The magnitude of the adjustment for the excess reactivity difference between integrals of moderator worth and rod worth is a function of the all rods banked critical height. As discussed in Reference 2, this correction is derived from measurements on the Core I mock-up assembly by comparing integrals of moderator worth and rod worth.

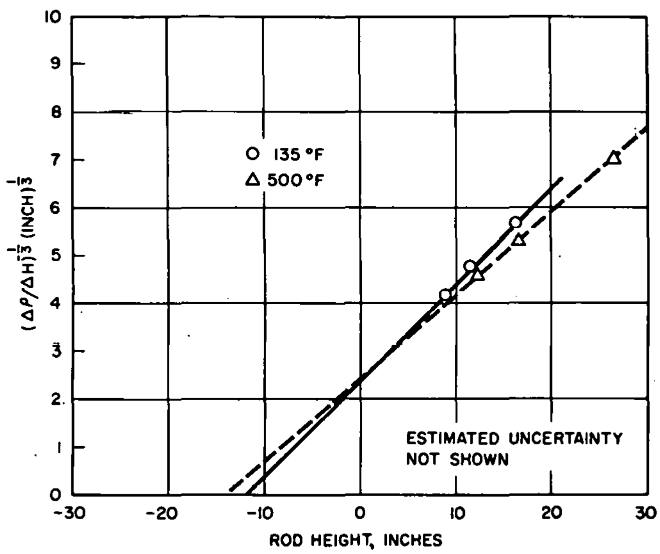

Figure II-5. Seed 2 Rod Bank Worth at Zero EFPH.

The method outlined above for calculating the excess reactivity using the inverse cube relationship has been verified by applying it to many types of core mock-ups studied in the PWR critical facility. However, in all of these cores, there existed uniform axial and azimuthal distribution of fuel and poison, a condition which does not exist in PWR during tests performed after power operation.

Departures from the linear relationship of the inverse cube root of rod worth as a function of rod height, which are observed for Core I, are attributed to non-uniform axial and azimuthal distributions of fuel and fission product poisons. The effects of these non-uniformities on rod worth values introduce uncertainties into the analytical calculation of the excess reactivity. The analytical method of determining the excess reactivity is useful only as long as the inverse cube relationship between rod worth and rod height remains linear. The measurements on Seed 1 after 2790 EFPH gave an indication of a departure from a linear relationship for the inverse cube root of rod worth. Since a similar behavior should also occur for Seed 2, the non-linearity of the inverse cube relationship of rod worth which was observed after 3560 EFPH on Seed 2 is evidence of non-uniform axial fuel depletion in the seed. The fact that this characteristic has not been so pronounced on Seed 2 as on Seed 1 may be associated with compensating non-uniform axial effects in the more highly irradiated blanket region for Seed 2.

When the inverse cubic relationship of the data is not valid, then the inverse cube method is no longer applicable and the excess reactivity must be determined graphically by numerical integration of the curves shown in Figure II-3 and II-4. from the all rods banked position to the fuel core height. Since experimental bank worth data are not available for critical rod heights high in the core until late in seed life, the curves in Figures II-3 and II-4 must be extrapolated to the full core height. Consequently, excess reactivity values determined by numerical integration of these rod worth curves are subject to considerable uncertainty.

The excess reactivity and temperature defect measurements obtained for the ambient and 500°F temperature conditions are shown in Table II-F. All excess reactivity values have been adjusted for the reactivity difference between integrals of moderator worth and rod worth as discussed in a prior paragraph. The temperature defect of reactivity is the difference between excess reactivity at ambient temperature and operating temperature. The decrease of the excess reactivity at 500°F with core age is shown in Figure II-6.

TABLE II-F

PWR-1 SEED 2 EXCESS REACTIVITY AND TEMPERATURE DEFECT MEASUREMENTS XENON-FREE AND APPROXIMATELY PEAK SAMARIUM CONDITIONS

		Ambient Condition		Hot Condition 500°F	
EFPH	Method	Temperature (°F)	Excess Reactivity (% $\Delta\rho$)	Excess Reactivity (% $\Delta\rho$)	Temperature Defect of Reactivity (% $\Delta\rho$)
0	Analytical *	135	16.00 ± 0.40	13.90 ± 0.10	2.10 ± 0.40
3560	Graphical †	140	11.8 ± 0.3	11.3 ± 0.4	0.5 ± 0.5
6145	Graphical	190	6.2 ± 0.9	5.5 ± 0.7	0.7 ± 1.0
7528	Graphical	130	4.6 ± 0.5	3.2 ± 0.5	1.4 ± 0.7

* Uncertainties are based on least square fitting of the data.

† Uncertainties are estimated on basis of extrapolation of rod worth curves.

Observation of excess reactivity and temperature defect values in Table II-F indicates irregularities in the results obtained at various times in seed life. The large uncertainties in results, particularly for the temperature defect, are indicative of the difficulty of measuring excess reactivities accurately on depleted cores having pronouced axial non-uniformities. The graphical method could be used with more confidence if additional rod configurations had been used for criticality at higher rod bank heights so that less extrapolation of the data would be required. No reactivity adjustment for the reactivity effect from samarium has been incorporated into the above results. The magnitude of the samarium reactivity effect is discussed subsequently in connection with a special test of samarium effects conducted near the end of Seed 2 life.

Temperature Coefficient of Reactivity Measurements

Temperature coefficients of reactivity are used (1) in the evaluation of reactor plant response and stability, (2) as corrections to results obtained during physics testing, and (3) as a quantity which may be compared with calculations to test the validity of calculational models.

Values of the temperature coefficient of reactivity at zero power are measured by the drift method during which the reactor moderator temperature is continuously increased over the range of temperature measurement. During this temperature drift, reactor periods are observed as a function of temperature for a given control rod bank height until criticality can no longer be maintained. The control rod bank is then repositioned to a new height which will re-establish a positive reactor period and the measurements are repeated.

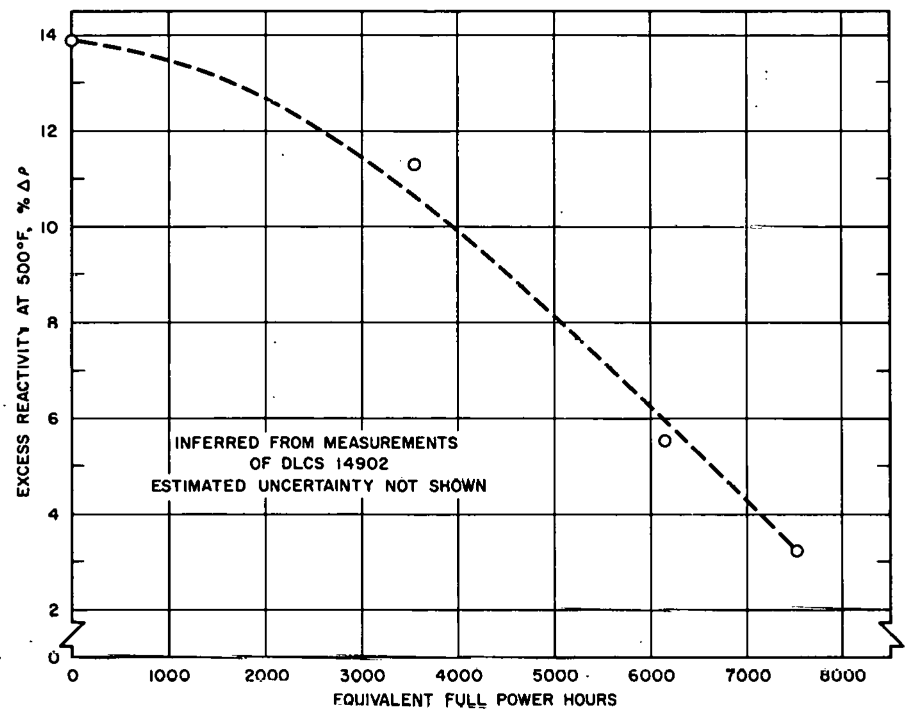

Figure II-6. Seed 2 Percent Excess Reactivity at 500°F vs Core Age.

For a major portion of Seed 2 life, the measurements of the zero power temperature coefficient were made with a rod configuration using the Group I and II rods banked, with the remaining rods fully inserted. This control rod pattern was selected because it was possible to achieve criticality with this rod configuration in the hot, xenon-free condition for the greater part of seed life. Late in seed life when criticality could not be achieved with this particular rod configuration, temperature coefficient measurements were made using an all rods banked configuration. The change in the rod bank configuration used may have affected the measured temperature coefficient values because of the influence of the changed rod configuration on the neutron leakage characteristics of the seed.

Since the maximum increment of reactivity change in an hour is approximately 40 x 10^{-4} $\Delta\rho$ at each rod bank height, it is possible to obtain several period measurements before the reactor becomes subcritical. From these period measurements, values of reactivity are inferred which give data for the variation of reactivity with temperature at each constant rod bank height. The experimental data are then fitted to linear relationships as a function of temperature by the method of least squares for each bank height. A slope determined in this manner by least square analysis is interpreted as the temperature coefficient at the average temperature of the data points.

In the temperature range from 235°F to 450°F, reactor heat is used to raise the moderator temperature such that temperature coefficient measurements at zero power cannot be obtained in this range. The need for using reactor heat arises from temperature limitations necessary on the running speeds of the primary coolant pumps. During heatup, reactor power is maintained at less than 1 percent of full power to eliminate the subsequent effects of transient xenon poisoning. Throughout the test, as the temperature is increased, coolant pressure is maintained constant with ±20 psi by bleeding water from the system and by using the pressurizer heaters.

Figure II-7 illustrates the temperature coefficient measurements as a function of temperature throughout Seed 2 life. It is observed from Figure II-7 that the temperature coefficient of reactivity at zero power shows a considerable decrease in absolute magnitude with core age. This decrease in the temperature coefficient with core age is similar to the behavior observed during Seed 1 operation.

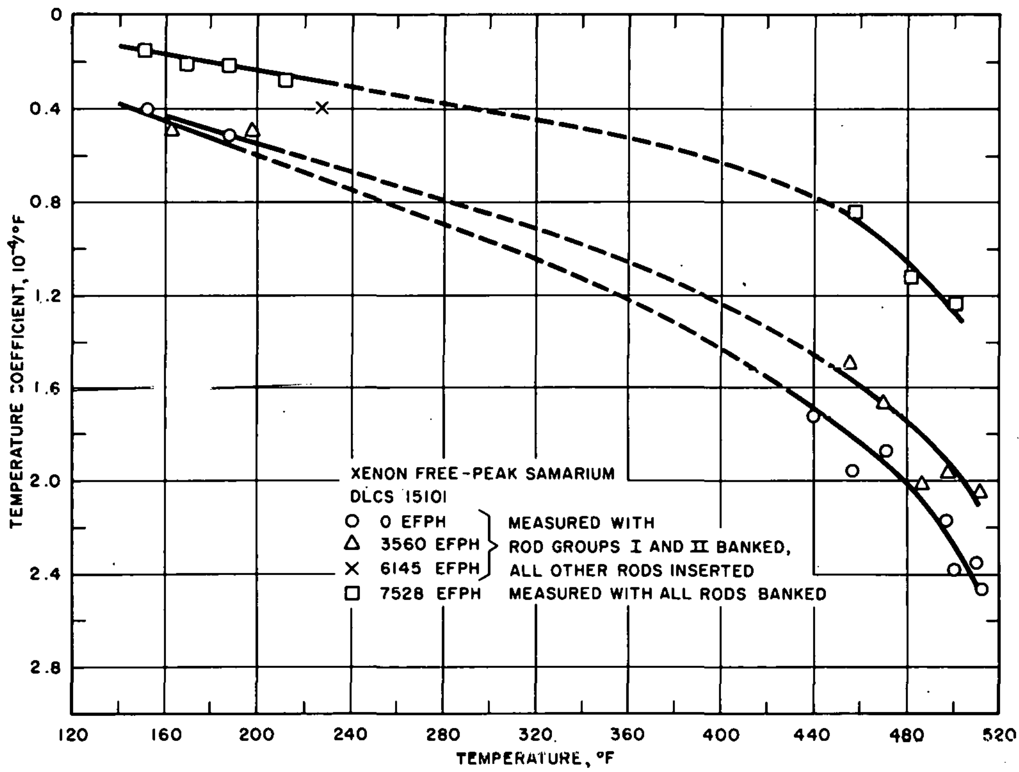

Figure II-7. Seed 2 Temperature Coefficient of Reactivity vs Temperature.

The temperature coefficient of reactivity was also measured periodically throughout Seed 2 life at full power under equilibrium xenon conditions. Table II-G contains a tabulation of all measurements of the temperature coefficient at full power during Seed 2 operating lifetime. The temperature coefficient at full power is inferred by determining the product of the controlling group rod worth ($\Delta\rho/\Delta h$) and the incremental change of rod height with temperature ($\Delta h/^{o}F$). Values of $\Delta h/^{o}F$ are determined from average readings of the Mueller Bridge inlet coolant temperature recorders for two different control rod bank heights sufficient to cause a temperature differential of several degrees at full power operation.

For each measurement of the full power temperature coefficient, the reactivity worth of the controlling rod group at a particular height must be known. Control rod reactivity worths cannot be measured at power; hence, the controlling group rod worth is measured only at zero power during a xenon transient. However, the rod worth at full power equilibrium xenon conditions can be approximated by the rod worth at the same height during xenon transient conditions. A condition of approximately similar axial distribution of xenon is expected at that time during a xenon transient at which the controlling rod group height is the same as the mean rod height measured at power for the $\Delta h/^{o}F$ measurement. Thus rod worths from xenon transients should be applicable except for minor differences due to coolant and fuel temperature distribution at power, and poisoning effects due to fission products.

It is not possible, however, to perform measurements of both the rod worth and $\Delta h/^{o}F$ at the same core age, and thus rod worth values are used for xenon transient measurements which most closely correspond in EFPH to the $\Delta h/^{o}F$ measurement. Some additional uncertainty is contained in the temperature coefficient at power measurements for those instances where considerable time has elapsed between the $\Delta h/^{o}F$ and $\Delta \rho/\Delta h$ measurements.

The temperature coefficient measurements at full power in Table II-G are plotted as a function of core age in Figure II-8. Also included in Figure II-8 are three values for the temperature coefficient at zero power, $500^{o}F$. As observed in Seed 1 operation, the temperature coefficient at full power is less negative than the zero power temperature coefficient. It is believed that this effect is attributable to the different leakage characteristics of the seed caused by the difference in total absorption at equilibrium xenon as compared with xenon free conditions, and to the different rod configurations used during the two types of measurements.

TABLE II-G

SEED 2 TEMPERATURE COEFFICIENT OF REACTIVITY AT FULL POWER

Controlling Rod Group	Rod Height (in.)	Time of $\Delta\rho/\Delta h$ Measurement (EFPH)	Rod Worth from Xenon Transient Measurements ($\Delta\rho/\Delta h$) (10^{-4}/in.)	Time of $\Delta h/^{o}F$ Measurement (EFPH)	$\Delta h/^{o}F$ Mueller Bridge (in./^{o}F)	Inferred Temperature Coefficient ($\Delta\rho/^{o}F$) ($10^{-4}/^{o}F$)
II	9.5	150	9.2 ± 0.5*	142	-0.219 ± 0.010*	-2.01 ± 0.14*
II	14.3	750	11.4 ± 0.5	717	-0.118 ± 0.010	-1.34 ± 0.13
II	16.3	750	13.0 ± 0.5	1120	-0.096 ± 0.005	-1.25 ± 0.08
II	18.8	1560	13.8 ± 0.5	1750	-0.098 ± 0.005	-1.35 ± 0.08
II	24.3	2240	16.0 ± 0.5	2210	-0.095 ± 0.003	-1.52 ± 0.07
II	33.0	2240	16.0 ± 0.5	3065	-0.106 ± 0.004	-1.70 ± 0.08
II	40.6	3560	16.0 ± 0.5	3520	-0.108 ± 0.007	-1.72 ± 0.12
II	65.3	4830	6.0 ± 0.5	4810	-0.283 ± 0.010	-1.70 ± 0.15
III	60.2	6145	5.3 ± 0.5	6115	-0.231 ± 0.010	-1.22 ± 0.13
IV	64.1	7528	4.2 ± 0.5	7514	-0.195 ± 0.008	-0.82 ± 0.10

* Uncertainties are estimated.

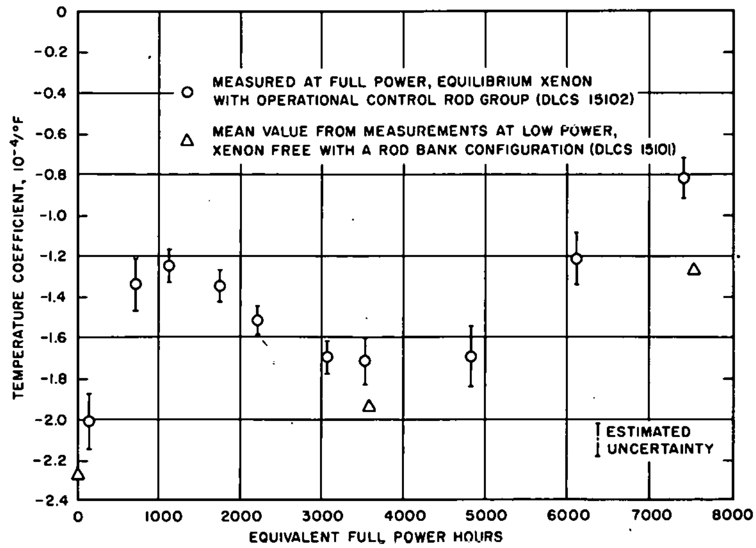

Figure II-8. Seed 2 Temperature Coefficient at 500°F vs Core Age.

A pronounced variation of the full power temperature coefficient for Seed 2 as a function of core age is indicated in Figure II-8. This behavior was not observed during Seed 1 operation because temperature coefficient measurements at full power were not begun until late in Seed 1 life. Specific reasons for the full power temperature coefficient variation remain unidentified beyond the observation that some variation with rod position can be anticipated because the temperature coefficient of reactivity is known to be sensitive to the neutron leakage properties of the core.

Xenon Transient Measurements

The xenon transient test following shutdown from full power equilibrium xenon conditions provides data for the determination of controlling group rod worths, control rod return time, peak xenon override time, and equilibrium and peak xenon critical rod positions. From these measurements, the equilibrium-to-peak xenon reactivity and average seed thermal flux level can be inferred. Each of the xenon transients was followed for approximately 40 hours after shutdown to provide data past the control rod return time. No data are obtained on the xenon-free condition of the reactor as a part of the test, but the xenon-free rod position and reactivity worth of the normal programming control rod configuration at operating temperature were presented in a previous section.

The performance of the xenon transient test provides a demonstration of the ability of the core to override transient xenon conditions at various times during lifetime. Although the maximum xenon lifetime was not measured directly, maximum xenon override capability is estimated to have existed up to approximately 5800 EFPH.

After shutdown from full power, the resulting reactivity transient caused by the buildup and decay of xenon is indicated by the critical positions of the controlling rod group as a function of time. Reactor period measurements which provide data for the determination of the critical rod height, return time, and controlling group rod worths are taken during the xenon transient. These quantities are obtained by the method of least square fitting of the experimental data of inferred reactivity values as a function of time.

Results derived from each of the xenon transient tests performed during seed lifetime are presented in Table II-H. The control rod return time is designated as that time after shutdown when the controlling rod group reaches the equilibrium xenon rod position and temperature recorded immediately before shutdown. From plots of the controlling group rod worth, the reactivity involved in going fr equilibrium to peak xenon concentrations is determined by numerical integration of the appropriate controlling group rod worth curves from the equilibrium xenon to peak xenon critical rod heights.

Figures II-9 through II-15 show typical plots of critical controlling group rod position as a function of time during a xenon transient for different times in seed lifetime. Controlling group rod worths measured during the xenon transient are shown in Figures II-16 through II-21. Figure II-16 is a composite plot of all Group II rod worth measurements. It is noted that each measurement in Figure II-16 tends to reach maximum worth values at successively higher axial locations in the core. This phenomenon, which is associated with changes in axial flux shape due to fuel depletion, is connected with the observed behavior of the temperature coefficient at power which is dependent on measured rod worth values. The measured values of Seed 2 rod worth are similar to those measured during corresponding Seed 1 tests so that no pronounced loss in rod worth is evident due to depletion of the neutron absorbing isotopes of the hafnium control rods.

TABLE II-H

SEED 2 XENON TRANSIENT MEASUREMENTS

EFPH	Control Rod Return Time (hrs)	Peak Xenon Override Time (hrs)	Equilibrium Xenon Critical Rod Position: Rod Group	Equilibrium Xenon Critical Rod Position: Rod Height (in.)	Peak Xenon Critical Rod Position: Rod Group	Peak Xenon Critical Rod Position: Rod Height (in.)	Equilibrium to Peak Xenon Reactivity (% $\Delta\rho$)	Average Seed Thermal Neutron Flux Level Inferred from Experimental Return Time (10^{13} n/cm^2 sec)
150	26.8	8.7	II	9.7	II	28.0	2.9	4.2*
750	28.2	8.8	II	14.8	II	37.3	3.5	4.7
1560	29.6	8.8	II	19.3	II	44.8	4.0	5.3
2240	31.7	9.3	II	24.9	II	51.0	4.2	6.1
3560	32.3	9.2	II	41.4	III	31.3	4.1	6.4
4830	31.9	8.7	II	67.4	IV	29.9	4.5	6.2
6145	32.8	---	III	62.3	Reactor Subcritical		---	6.7
7528	34.4	---	IV	71.1	Reactor Subcritical		---	7.5

* Assumes a microscopic thermal absorption cross section of 1.938×10^{-18} cm^2 for Xenon-135

Results obtained from inferred reactivity values corrected to 500°F, 1800 psia.

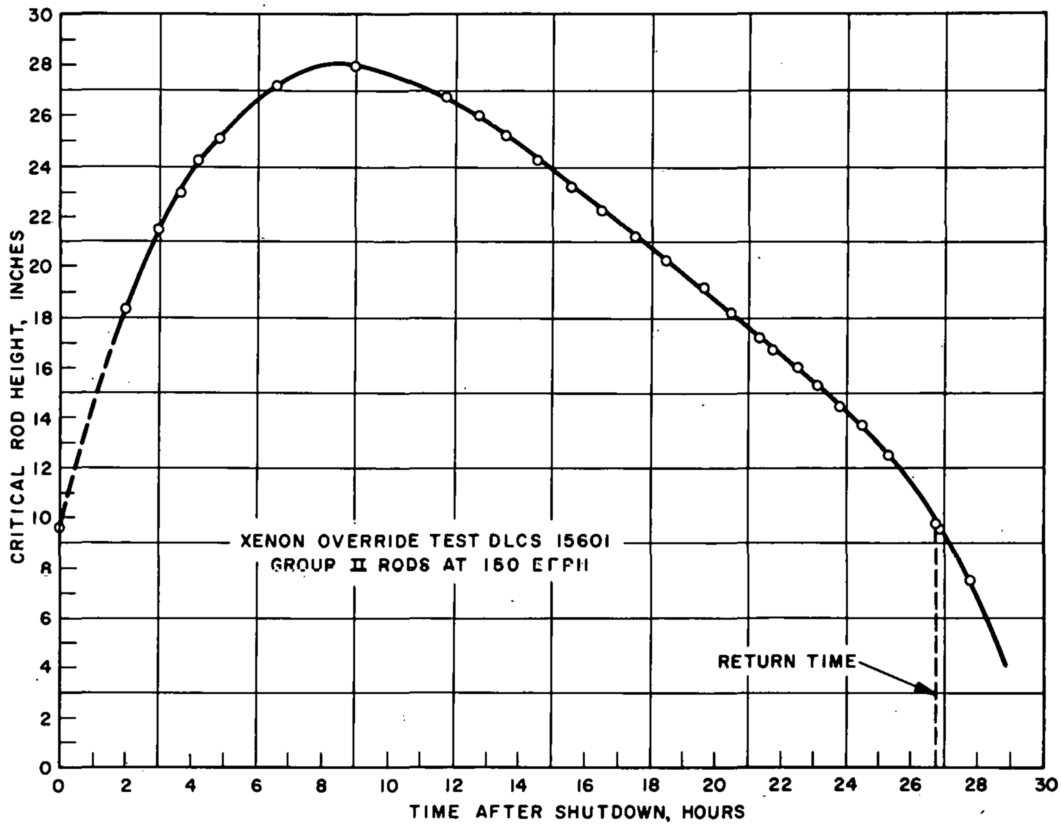

Figure II-9. Seed 2 Critical Rod Group Height vs Time after Shutdown (150 EFPH).

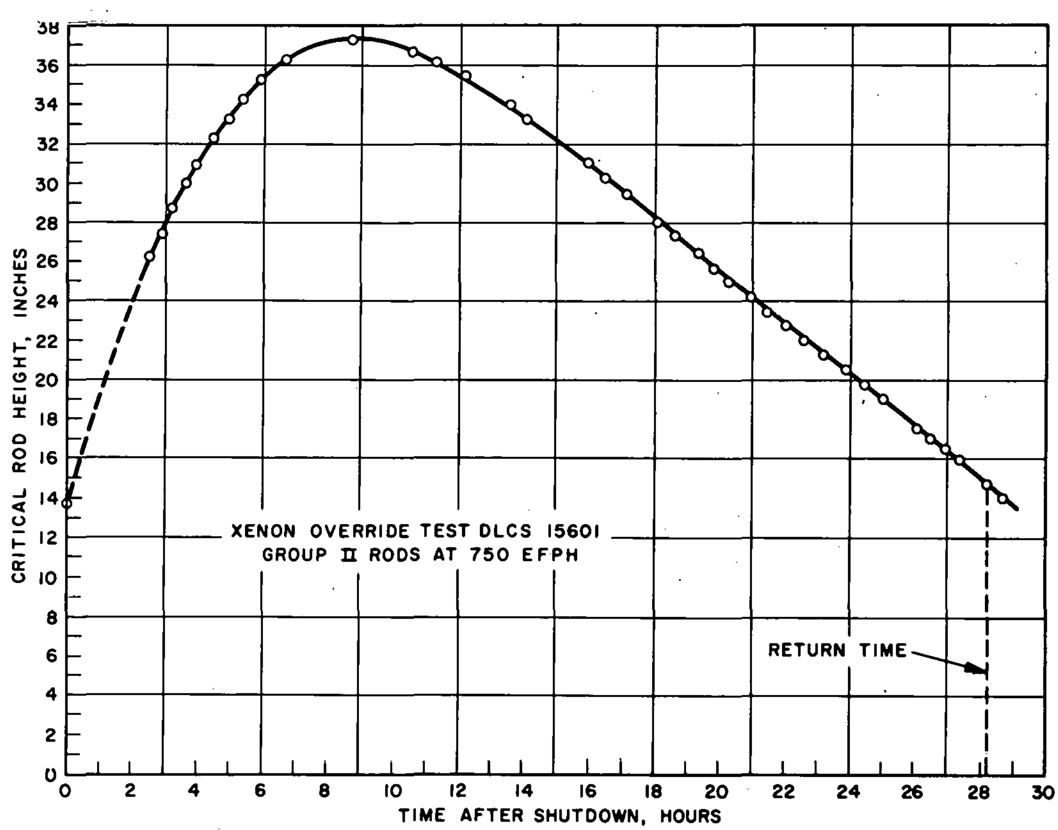

Figure II-10. Seed 2 Critical Rod Group Height vs Time after Shutdown (750 EFPH).

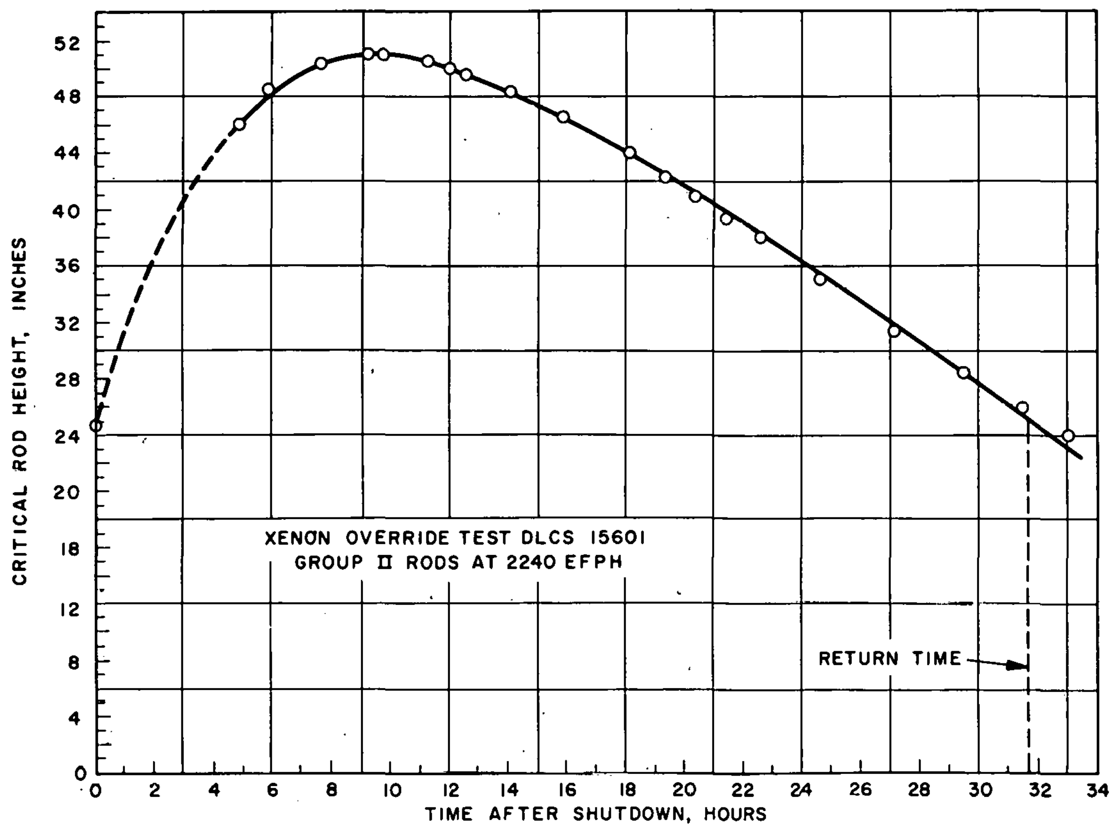

Figure II-11. Seed 2 Critical Rod Group Height vs Time after Shutdown (2240 EFPH).

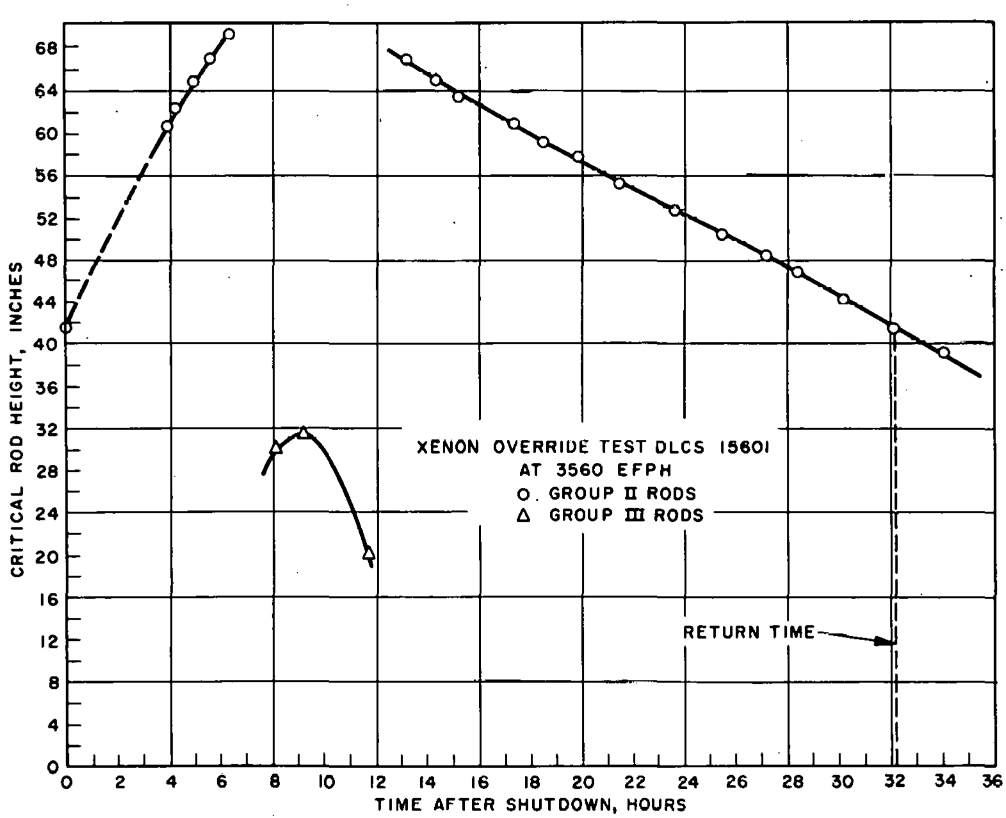

Figure II-12. Seed 2 Critical Rod Group Height vs Time after Shutdown (3560 EFPH).

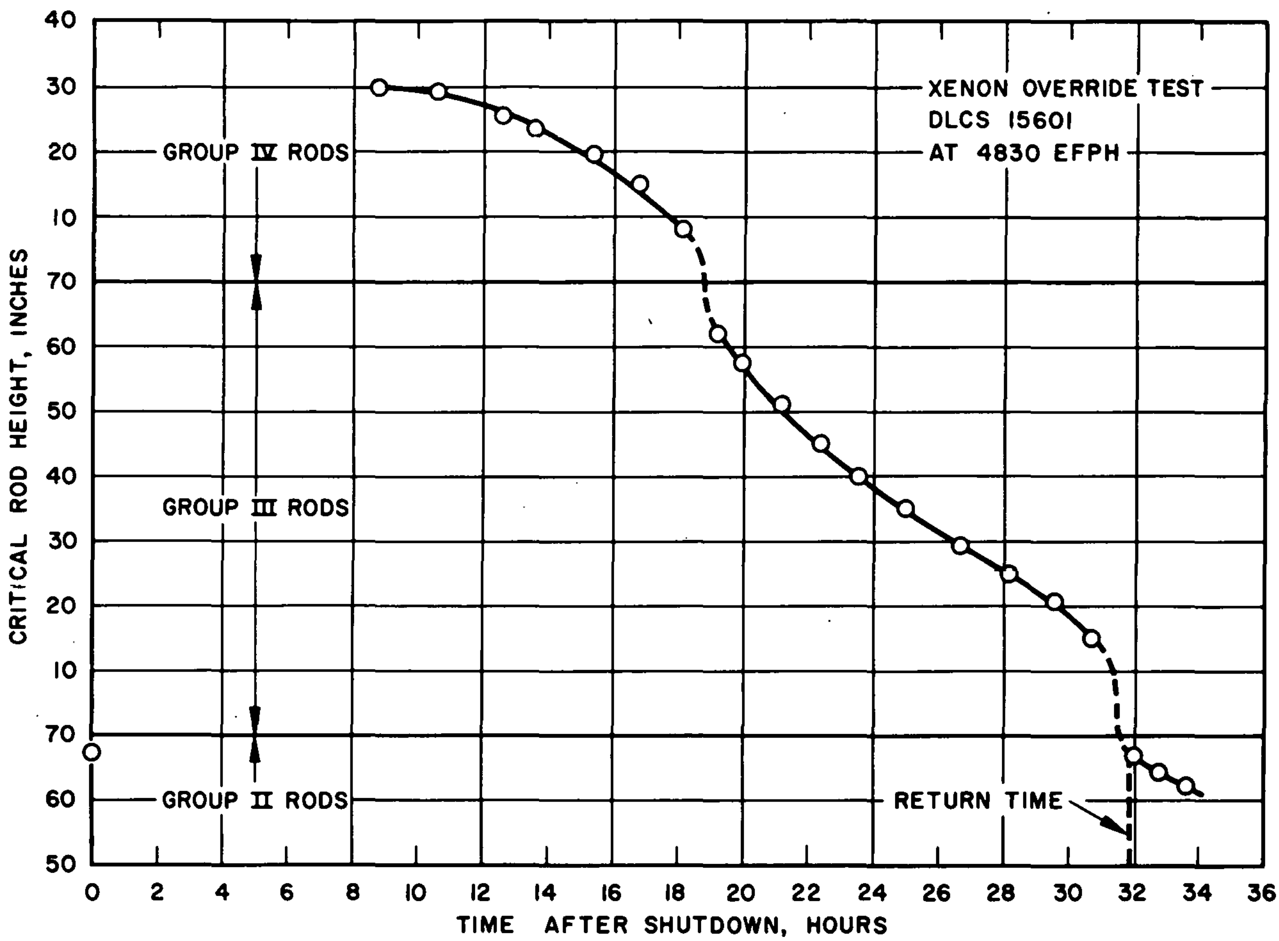

Figure II-13. Seed 2 Critical Rod Group Height vs Time after Shutdown (4830 EFPH).

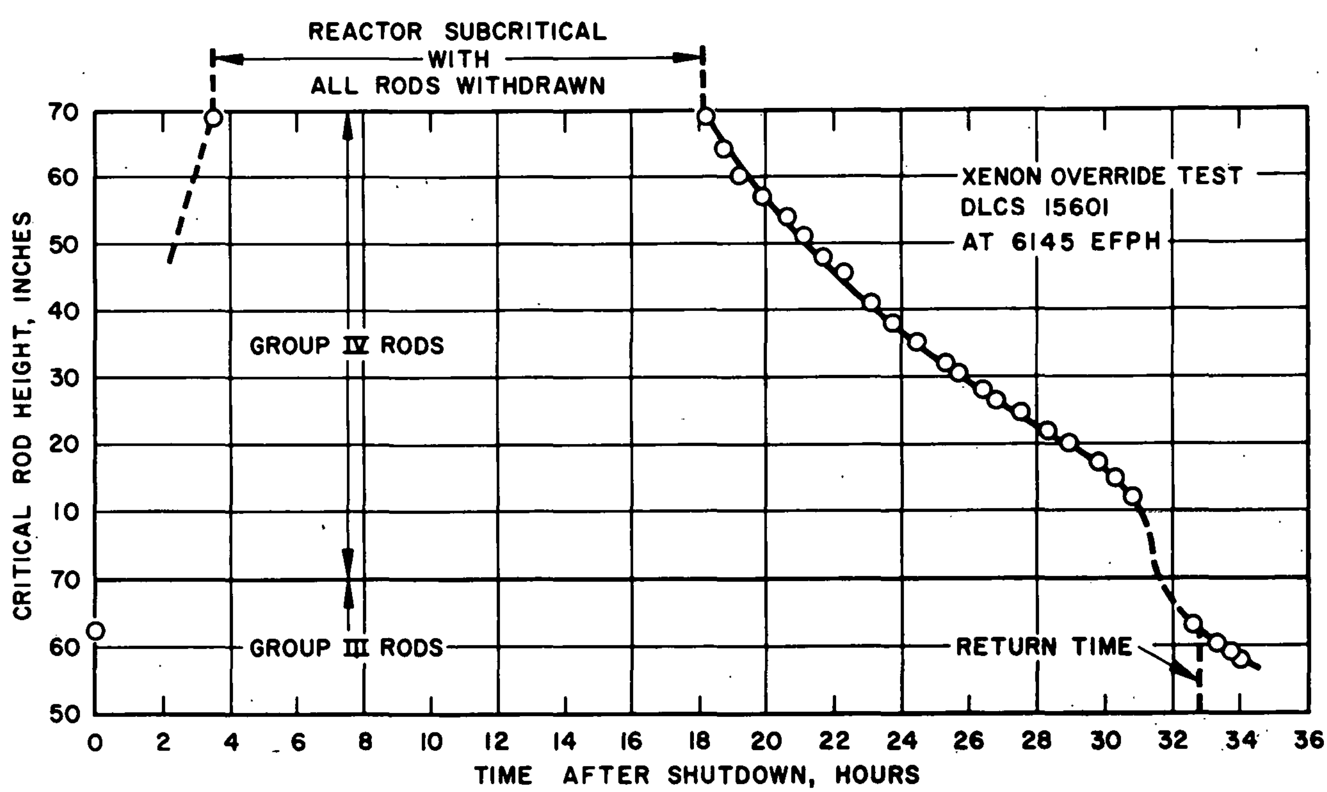

Figure II-14. Seed 2 Critical Rod Group Height vs Time after Shutdown (6145 EFPH.

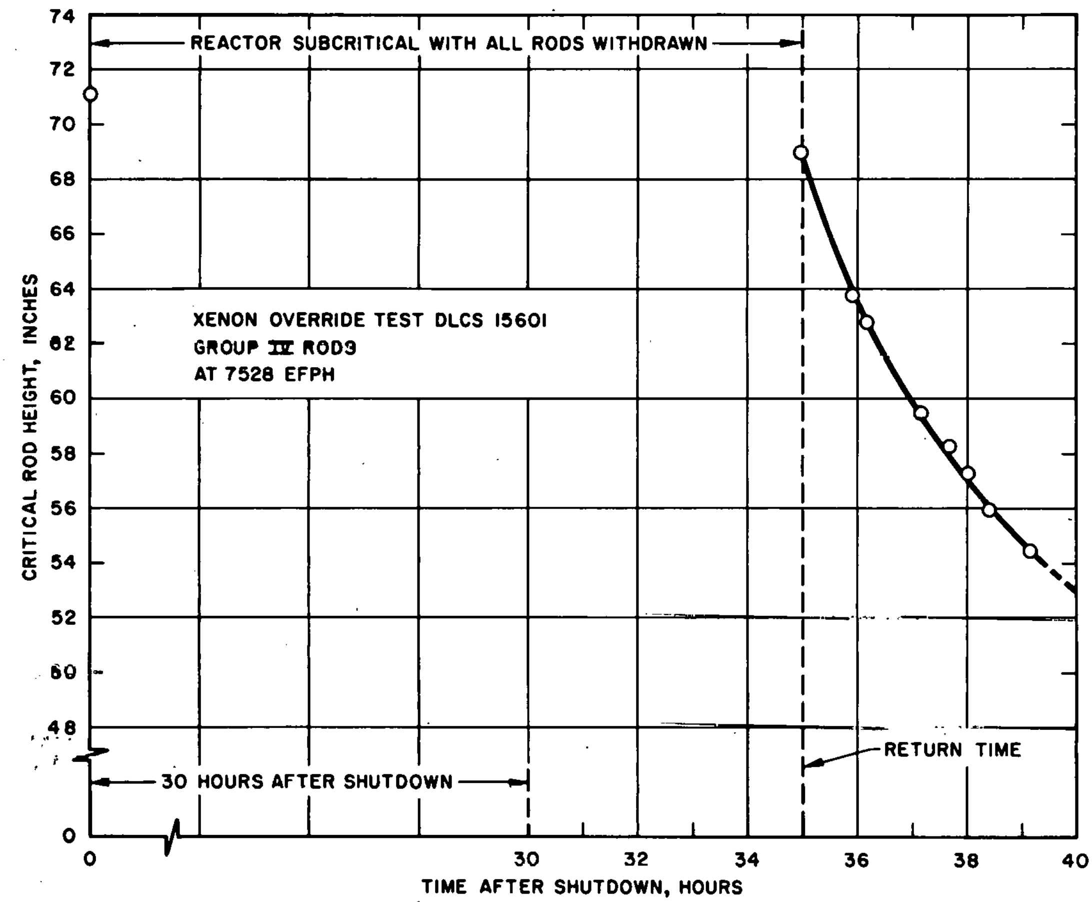

Figure II-15. Seed 2 Critical Rod Group Height vs Time after Shutdown (7528 EFPH).

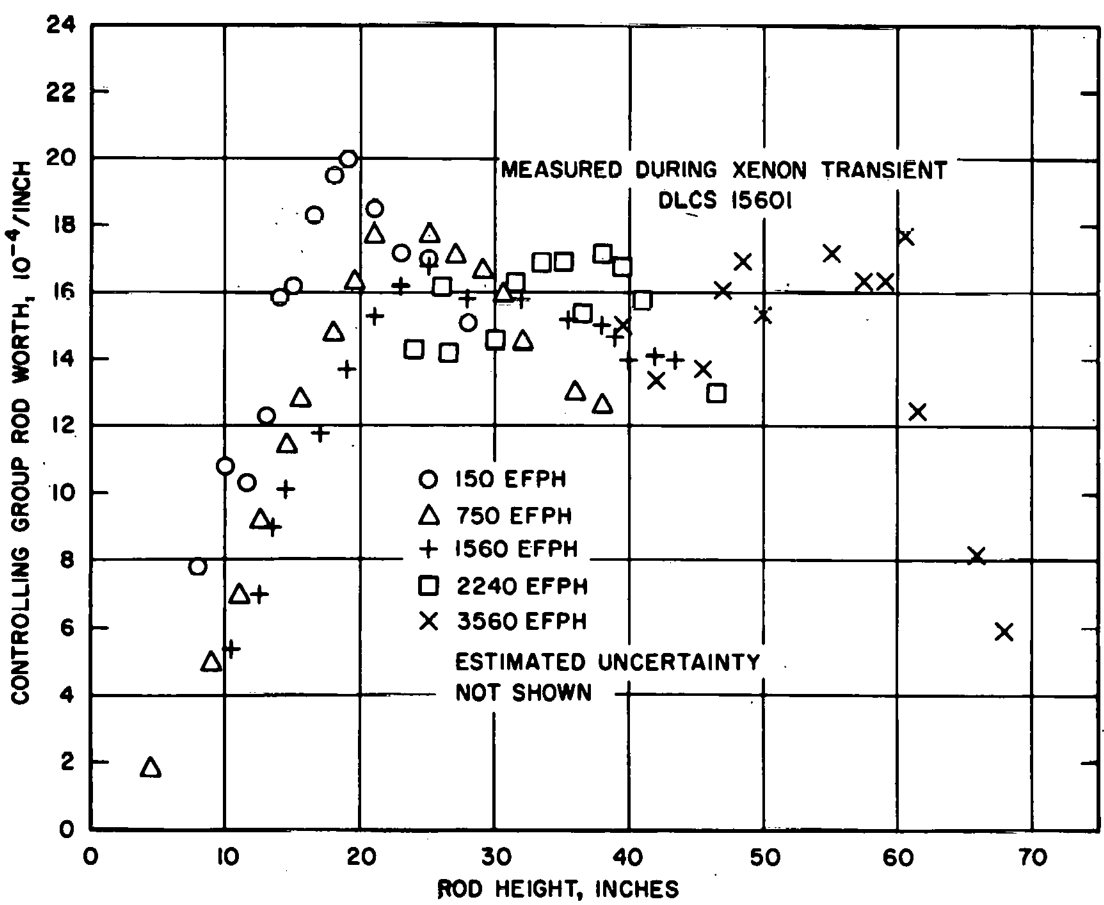

Figure II-16. Seed 2 Group II Rod Worth vs Rod Height.

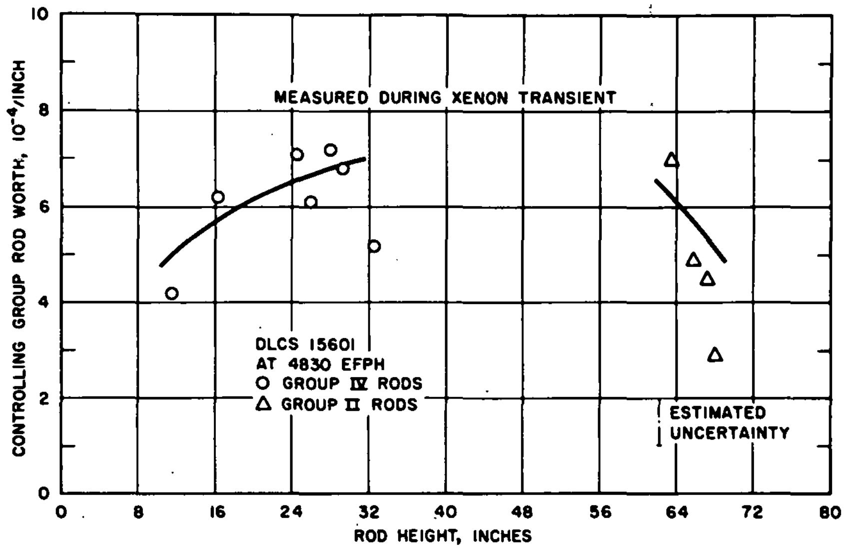

Figure II-17. Seed 2 Rod Worth vs Rod Height.

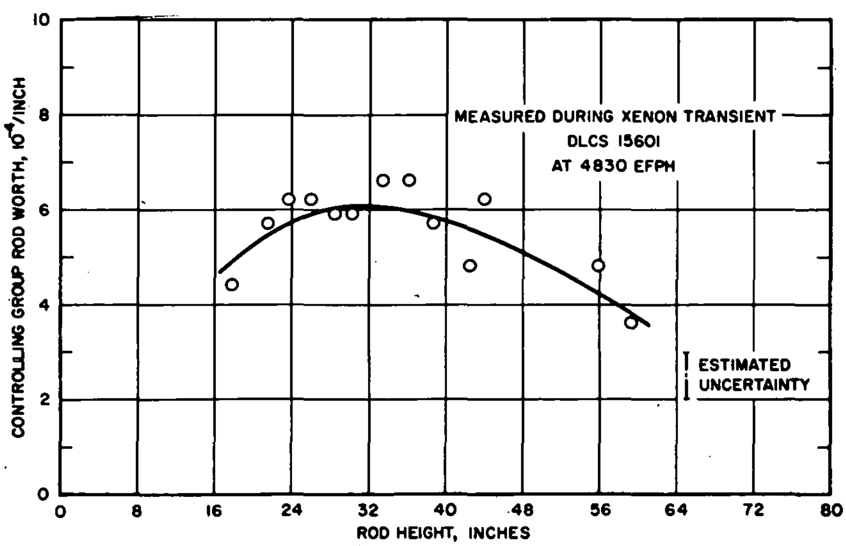

Figure II-18. Seed 2 Group III Rod Worth vs Rod Height (4830 EFPH).

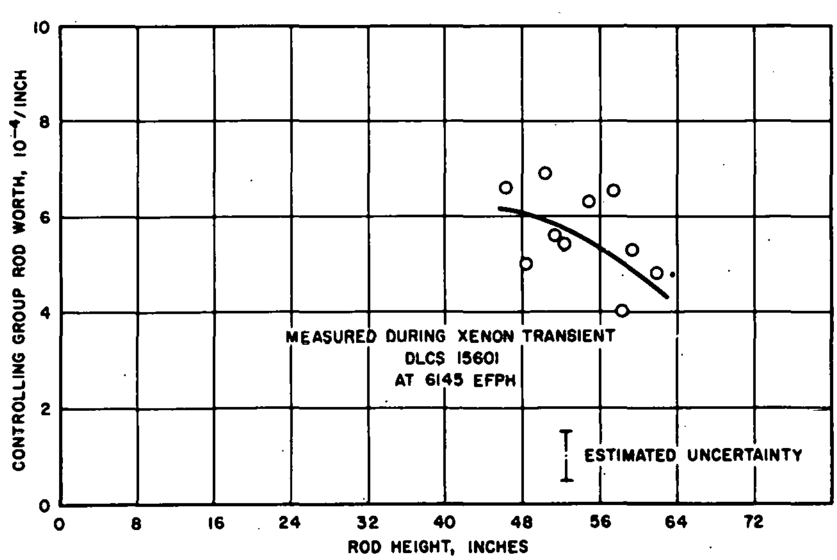

Figure II-19. Seed 2 Group III Rod Worth vs Rod Height (6145 EFPH).

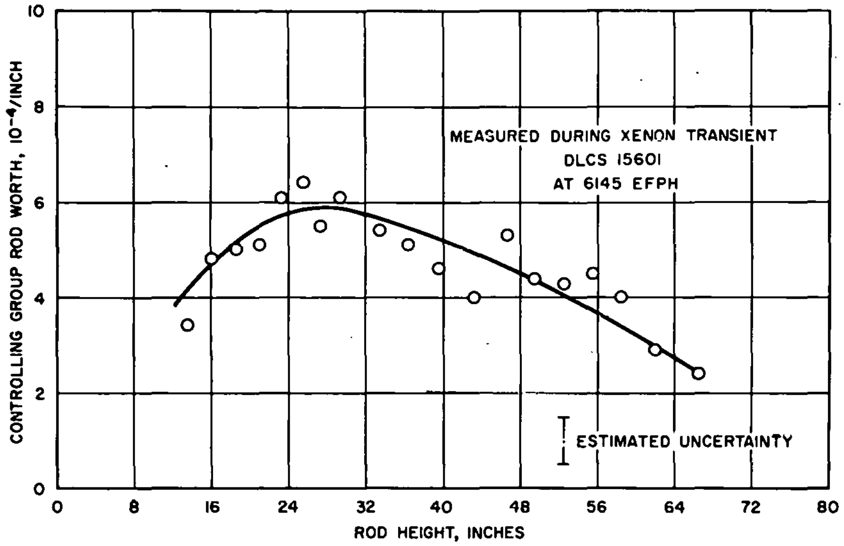

Figure II-20. Seed 2 Group IV Rod Worth vs Rod Height (6145 EFPH).

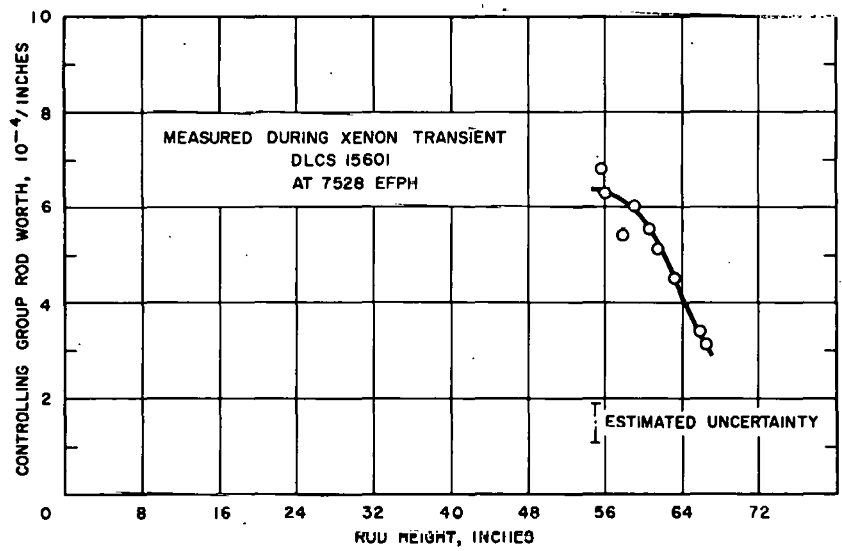

Figure II-21. Seed 2 Group IV Rod Worth vs Rod Height (7528 EFPH).

Average Seed Thermal Flux Level

The average seed thermal flux level is calculated using the experimentally determined control rod return time. Solution of the time dependent xenon concentration equations for a point reactor yields an expression for the average thermal neutron flux given by:

$$\Phi = \frac{(\nu_I + \nu_X)(\lambda_X - \lambda_I)(1 - e^{-\lambda_X T})}{\sigma_X \nu_I (e^{-\lambda_I T} - e^{-\lambda_X T})} - \frac{\lambda_X}{\sigma_X} \qquad (1)$$

where

ν_I, ν_X = iodine and xenon fission yield fractions,

λ_I, λ_X = iodine and xenon decay constants,

σ_X = xenon microscopic cross section, and

T = time after shutdown required for the control rods to return to their equilibrium xenon position.

Each of the above quantities is constant except the return time, T, and the xenon cross section σ_X, which change with core age.

Values for the average seed flux calculated from equation (1) using the experimentally determined control rod return time are given in Table II-H for various times in seed life. A plot of the inferred seed flux levels during Seed 2 life is shown in Figure II-22. Average seed thermal flux levels inferred in this manner for Seed 2 have been generally lower than the corresponding Seed 1 flux values. The reduction in average flux levels for Seed 2 is expected because of the higher fuel loading of Seed 2 over Seed 1 such that the same power level can be maintained with a lower flux level.

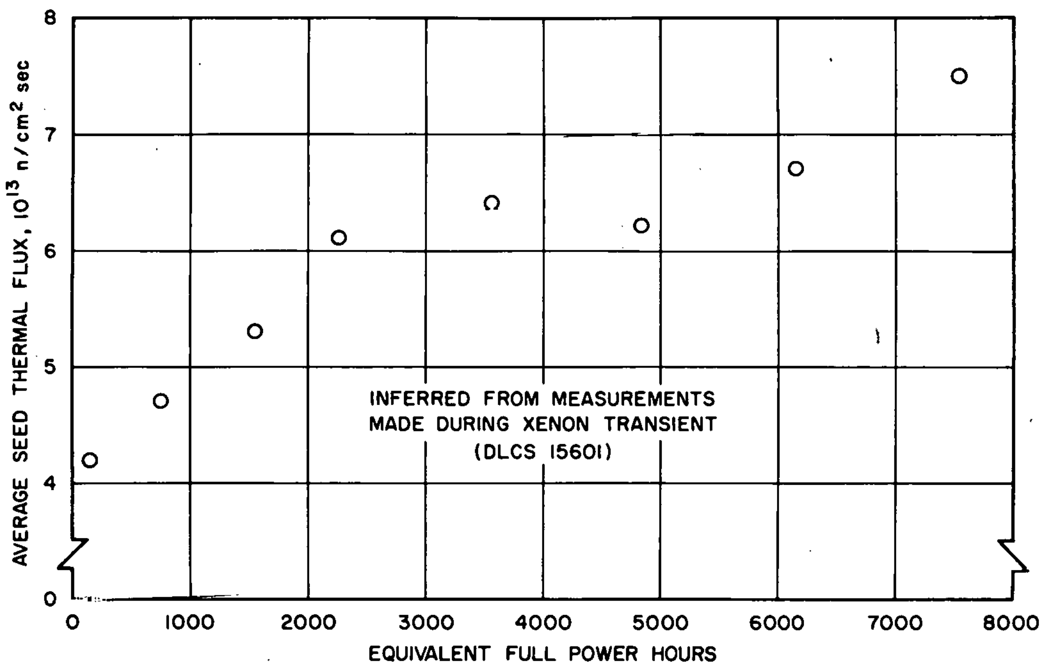

Figure II-22. Seed 2 Average Seed Thermal Flux Level vs Core Age.

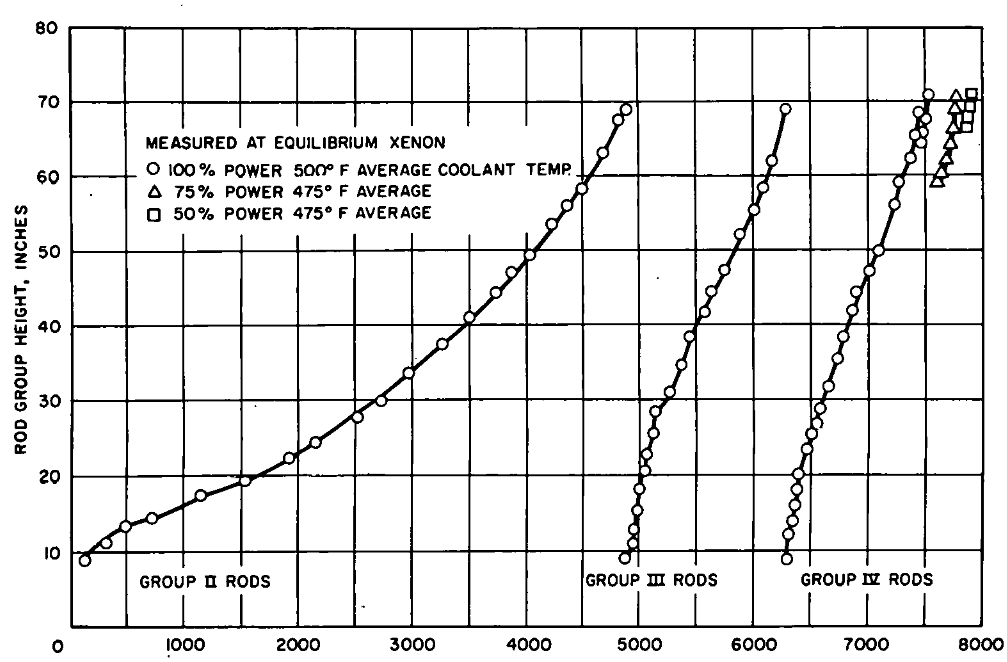

Figure II-23. Seed 2 Controlling Rod Group Height vs Core Age.

Reactivity Lifetime Measurements

A plot of the position of the controlling rod group height at equilibrium xenon as a function of Seed 2 lifetime is shown in Figure II-23. The end of Seed 2 full power equilibrium xenon lifetime occurred in mid June 1961 when all control rods were withdrawn to 71 inches and the average moderator temperature reached 497°F. A total of 7528 EFPH was accumulated on Seed 2 at rated full power equilibrium xenon conditions.

The energy utilization of the core was extended beyond the full power lifetime by operation at reduced power levels and with an average moderator temperature of 475°F. The reactivity gained from the temperature defect between 475°F and 500°F and the smaller xenon concentration at reduced power levels were sufficient to extend the life of the seed an additional 372 EFPH. Thus a total operating lifetime of 7900 EFPH was accumulated on Seed 2.

An indication of the transient samarium effect on controlling rod height was seen after resuming power operation at the 50 percent power level. The Group IV rod height increased to 65.7 inches 58 hours after start-up then gradually decreased to 64.9 inches about 112 hours after startup, and increased thereafter. Also during the 50 percent power run, a small reduction in power level was attempted in an effort to extend reactivity lifetime. However, a xenon transient was introduced which required the rods to be moved toward the upper limit at an accelerated rate. The power level was then increased to avoid further rod withdrawal to the full out position which would have caused premature subcriticality. It was found that changes in power level had considerable effect on the reactivity behavior of the core during reduced power operation.

Fluctuations in the controlling rod group position as a function of core age were observed for intervals during full power operation following shutdown for physics testing at 4800 and 6100 EFPH on Seed 2. These fluctuations occurred first when the Group III, and also when the Group IV rods were controlling low in the core. Examination of the data recorded from the seed metal thermocouples during the periods of unusual rod motion suggested the presence of a small axial power fluctuation. In an effort to determine whether this phenomenon is related to a tendency for fluctuations of the Xenon 135 spatial distribution, axial and radial flux perturbation tests were performed. The above control rod and Xenon 135 spatial distribution variations are discussed subsequently in Chapter 4, Special Seed 2 Physics Tests.

Another core parameter which is measured during full power operation by means of flow rate and thermocouple data is the seed-blanket power sharing fraction. This parameter is of interest because of the direct relationship it has to core lifetime, since the larger the share of power from the blanket the longer the seed lifetime will be. Measured and calculated seed power fraction values obtained for various times in Seed 2 life are shown in Figure II-24. The measured values, which have an uncertainty estimated at one percent in core power, exhibit a scatter which may be, in part, associated with the fact that three or four coolant loops were operative during the measurements as indicated in Figure II-24. No reason for the apparent effect of the number of loops on seed power fraction has been identified since the measurements take into account the different flow patterns with three or four loops operating.

Using two different calculational models, the curves shown in Figure II-24 were obtained for the power sharing behavior during Seed 2 life. A one-dimensional depletion calculation (CANDLE) appears to overestimate the seed power fraction relative to the measured data. Conversely, the three-dimensional (DRACO) analysis gives seed power fractions smaller than the measurements indicate.

In contrast to both calculations, the measurements provide no indication of an increase in the average seed power fraction for Seed 2 over that for Seed 1 (Reference 2). This difference favors a longer Seed 2 reactivity lifetime and a lesser actual reactivity loss rate with EFPH, relative to Seed 1, than calculations had predicted. The 7900 EFPH lifetime actually attained for Seed 2 exceeded expectations based on the above calculations by some 1000 EFPH. This phenomenon may be associated with a reactivity behavior of the natural uranium blanket fuel which is more favorable than calculations indicated.

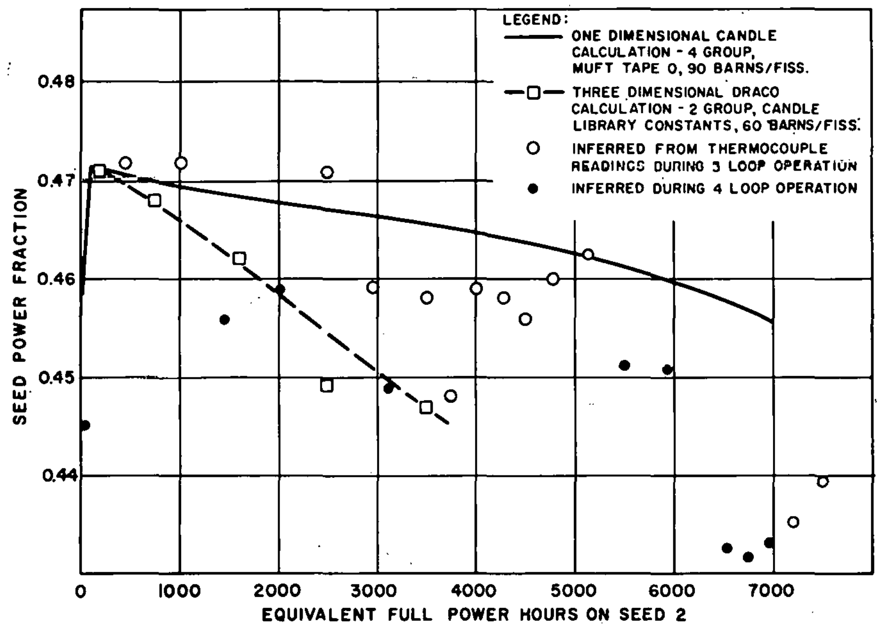

Figure II-24. Seed 2 Power Sharing vs EFPH.

<u>End of Life Testing</u>

Following the end of full power capability, critical rod bank height and worth measurements were made to examine the reactivity effects of the relative fuel depletion and fission product poisoning in symmetrical sections of the core. Using symmetrical patterns of adjacent rods a minimum number of adjacent control rods in a section of the core were withdrawn to achieve a critical rod bank height. Critical bank height and reactivity worth measurements were made for control rod configurations symmetric about each of the number 8 inset corner rods and the number 4 center flat control rods. The corner measurements were performed using a five rod bank while face measurements used a nine rod bank. The results of the core reactivity symmetry measurements at the end of Seed 2 full power lifetime are given in Table II-1. Figure II-2 illustrates the location and identification of individual control rods used for the core symmetry measurements.

Comparison of the relative corner reactivities obtained at end of Seed 2 life with those obtained at the start of life (Table II-D) indicates that the four corners are not in the same relative order as in the beginning of life. However, no particular significance is attributed to this apparent shift in the asymmetry because of the small reactivity values involved at the beginning of life and particularly at the end of life. Such small differences may arise from uncertainties in measuring absolute control rod positions. At the beginning of Seed 2 life, addition of a new blanket assembly in core location F-2, and new blanket bundles in E-6 and H-3 apparently did not alter the relative reactivity asymmetry of the core as measured at the end of Seed 1. No noticeable increase in rod worth can be attributed to the insertion of a new hafnium control rod in location 82. Therefore, it is concluded that no appreciable power asymmetry occurred during Seed 2 operation which would have been evidenced as a reactivity asymmetry at end of life.

TABLE II-1

SEED 2 END OF LIFE REACTIVITY SYMMETRY MEASUREMENTS

Location	Control Rods in Bank	Symmetrical about Rod Number	Critical Rod Bank Height H_c (in.)	Rod Bank Worth ($\Delta\rho/\Delta h$) (10^{-4}/in.)	Relative Reactivity (% $\Delta\rho$)
Northwest corner	64, 74, 81, 11, 21	81	66.56 *	26.03 $\pm$ 0.52	0
Northeast corner	61, 71, 82, 12, 22	82	66.78	24.59 $\pm$ 0.49	-0.054
Southeast corner	62, 72, 83, 13, 23	83	66.40	25.38 $\pm$ 0.77	0.041
Southwest corner	63, 73, 84, 14, 24	84	66.85	24.21 $\pm$ 0.48	-0.070
North face	81, 11, 21, 31, 41, 51, 61, 71, 82	41	63.15	24.10 $\pm$ 0.48	0
East face	82, 12, 22, 32, 42 52, 62, 72, 83	42	63.00	21.88 $\pm$ 0.61	0.033
South face	83, 13, 23, 33, 43 53, 63, 73, 84	43	63.22	24.24 $\pm$ 1.95	-0.017
West face	84, 14, 24, 34, 44, 54, 64, 74, 81	44	63.16	22.88 $\pm$ 0.46	-0.002

* The uncertainty of absolute rod heights is estimated to be $\pm$0.25 inch.

CHAPTER 4

SPECIAL SEED 2 PHYSICS TESTS

During the course of Seed 2 life, three special purpose physics tests were performed to investigate certain nuclear characteristics of Core I which are not readily observable from the normal periodic tests. Two of these tests were intended to explore the susceptibility of the core for fluctuations in the spatial power distribution which result from flux and xenon perturbations induced by prescribed abnormal control rod movements. The Axial Flux Perturbation test (DLCS 38801) and the Radial Flux Perturbation Test (DLCS 38802) were conducted after periodic fluctuations were observed in the controlling rod group height as a function of time at full power. The Samarium Transient Test (DLCS 15604) was conducted to observe the time dependent reactivity effect from buildup of peak samarium concentration following shutdown.

Rod Position Fluctuation During Normal Operation

Following resumption of full power operation after shutdown for zero power physics testing at 4830 and 6145 EFPH, periodic fluctuations were observed in the height of the Group III and IV control rods, respectively, as a function of time. The data from the various core instruments during these periods of control rod fluctuations were examined for indications of the source of the disturbance. The behavior of various reactor plant parameters is illustrated as a function of time in Figure II-25 and Figure II-26 for the two intervals of interest following the reactor shutdowns mentioned above.

On both occasions, departures of several inches from the expected control rod positions are evident from the figures. In both instances, the controlling rod positions stabilized after the rod group had been substantially withdrawn. Each of these occurrences lasted several days; the first began with the Group III rods controlling on February 6, 1961, and the second with the Group IV rods controlling on April 19, 1961.

Reactor plant parameters displayed in Figures II-25 and II-26 include the inlet coolant temperature, T_c, outlet coolant temperature, T_h, and the average of the inlet and outlet temperature, T_{avg}. Individual nuclear instrumentation channel readings calibrated in terms of percent of full power are also indicated. Locations of the nuclear instruments are shown in Figure II-2.

Data from the seed fuel thermocouples are shown in Figures II-25 and II-26 for selected seed cluster locations. Individual core cluster locations are shown in Figure II-2 and are identified by alphabetical (vertical) and numerical (horizontal) coordinates. These selected seed clusters are each equipped with four thermocouples positioned in the center of a fuel plate at 5.75, 19.5, 34.5, and 51.0 inches from the bottom of the cluster. These thermocouples thus measure the axial variation of the centerline temperature of the fuel plate. Experience indicates that for a valid indication of a temperature rise or fall, the thermocouple readings should vary by more than several degrees from a nominal value. Slight variations of only a degree or two are not necessarily indicative of a temperature change but result from drift and error of the recording instrument.

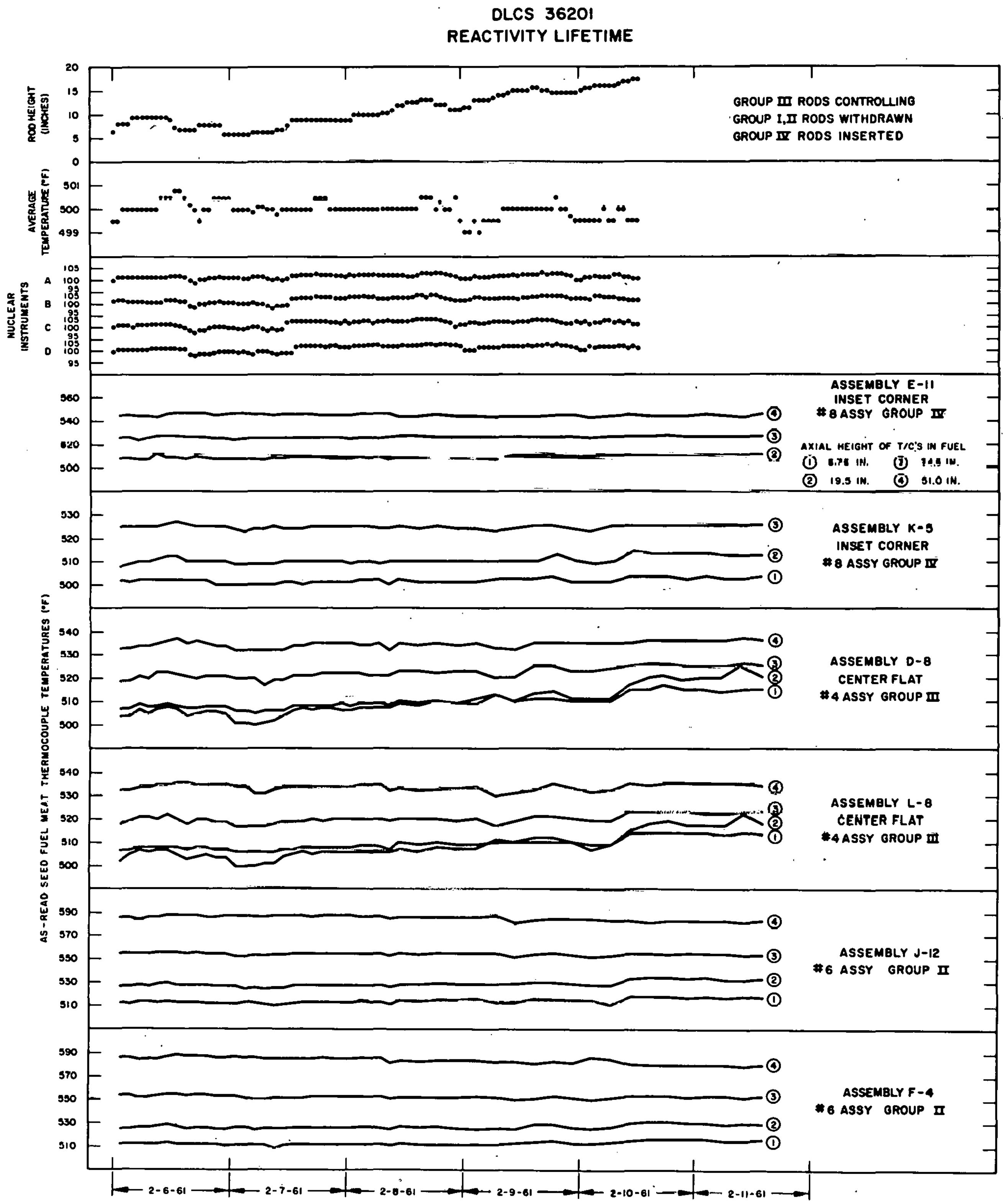

Figure II-25. Seed 2 Reactor Instrument Readings vs Time.

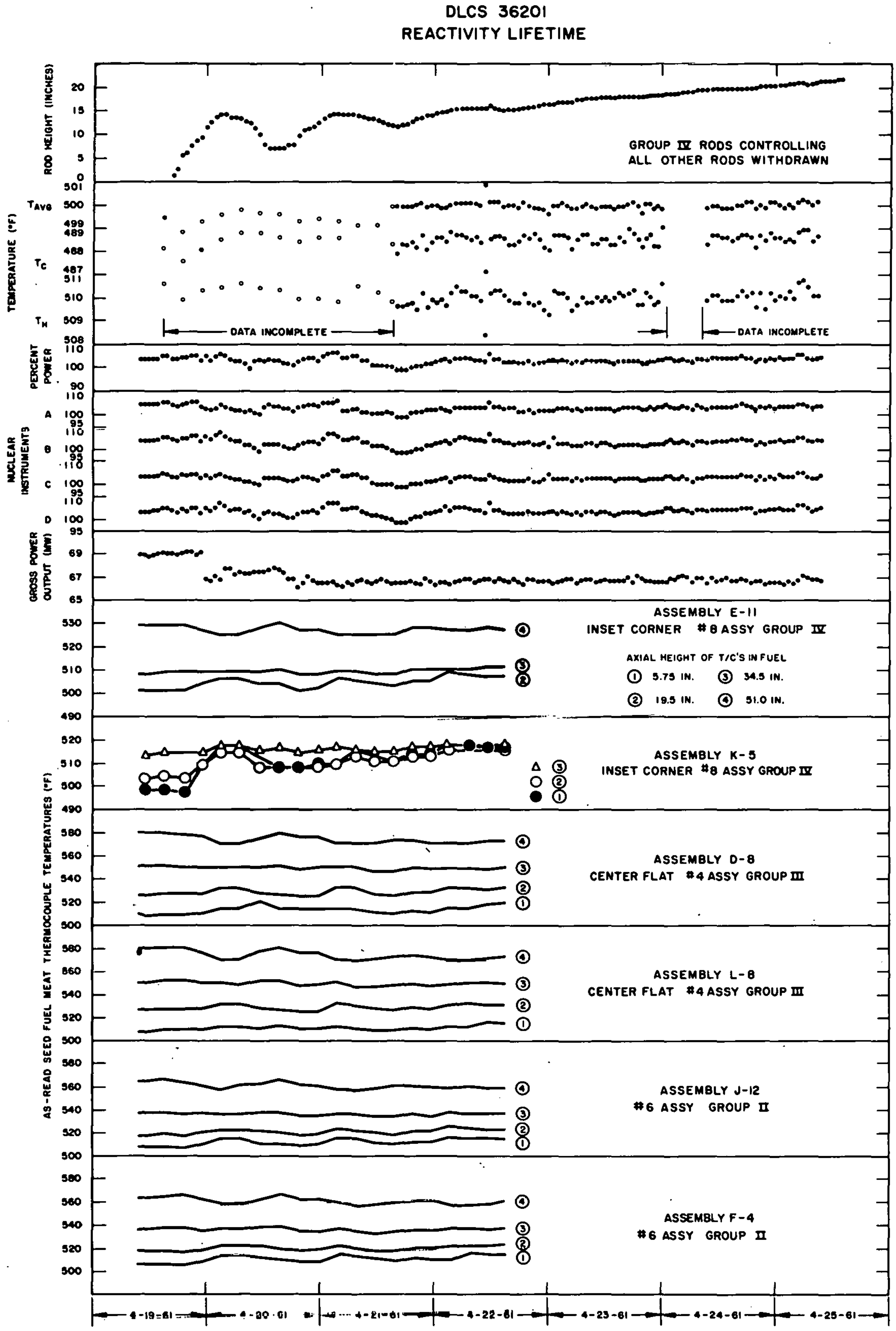

Figure II-26. Seed 2 Reactor Instrument Readings vs Time.

Figures II-25 and II-26 reveal that, while the control rods underwent unusual fluctuations, none of the nuclear instruments indicated any variations which suggest that operation of the reactor was abnormal. However, examination of the data from the seed fuel thermocouples gives some indication of a disturbance in the power distribution. Axial power distribution changes could be shown by the opposing behavior of two thermocouple positions. For example, if the top termocouple position temperature decreased while a lower position temperature increased concurrently, a shift in the power distribution towards the bottom of the core would be indicated. Thermal analysis shows that for Core I the avial fuel plate temperature is a combined function of the integrated temperature rise in the water channel and of the temperature gradient across a fuel plate. The water channel temperature rise is related to the integrated power of the fuel plate from the bottom of the plate to the point in question; the fuel plate temperature gradient is related to the pointwise power in the fuel plate. The centerline fuel plate temperature is thus influenced by several parameters: radial and axial position in the core, EFPH, rod position, coolant flow, and number of coolant loops. However, a dominating factor determining the centerline temperature is the axial flux shape. The axial flux shape is most sensitive to change when the control rods are controlling in the lower portion of the core. Thus a perturbation of the axial power distribution low in the core would be reflected in a compensating change in the upper portion of the core in order to maintain the integral of total power constant.

Such a change in the axial flux distribution would occur when criticality control of the reactor is transferred from a rod group that is fully withdrawn to one that is fully inserted. When the axial power distribution changes rapidly during power operation, a delayed but significant change would occur in the distribution of neutron-absorbing Xenon 135. This delayed effect of the Xenon 135 is a result of its formation from the decay of Iodine 135, which is a principal fission fragment. Thus the equilibrium between the axial power distribution and the Xenon 135 poison distribution is disrupted by the initial withdrawal of a fully inserted rod group, thereby causing a period of readjustment between the axial distribution of power and Xenon 135. This phenomenon results in fluctuation in the controlling rod height required to maintain criticality while the Xenon 135 poison shifts around in the core.

Axial Flux Perturbation Test

After observing the two occasions of control rod fluctuations, further investigation was conducted to examine the core conditions which could lead to unbalanced power and xenon distributions. Efforts were directed towards determining reactor behavior when the Xenon 135 fission product poison distribution is altered. The axial flux perturbation test was therefore conducted in an attempt to force control rod fluctuations by rapidly changing the axial power and xenon distributions.

The basic test method was to produce an axial non-equilibrium xenon distribution by rapidly altering the position of control rods. Since this test was performed late in Seed 2 life with the controlling Group IV rods withdrawn to above 50 inches, little latitude was available for selection of the rod groups to be repositioned. This test was performed at a 90 percent power level, commencing at approximately 7160 EFPH. The reactor overall power level was reduced to 90 percent to provide a margin of safety for reactor protection in the event that substantial power redistribution should develop from the respositioning of the control rods.

Prior to changing the position of the control rods, reactor power was kept constant at 90 percent for approximately 40 hours to establish an equilibrium distribution of xenon in the core. A non-equilibrium xenon distribution was sought by partial insertion of all rods at the upper programmed limit and withdrawal of the controlling Group IV rods to maintain power level. The amount of insertion of the previously fully withdrawn Group I, II, and III rods was such that the Group IV control rods were withdrawn 4.9 inches. Further repositioning was not undertaken in order to avoid exceeding a limitation placed on the exit water temperatures from the seed clusters. This limitation was established on the basis of readings obtained prior to the test so that a known safe thermal condition would not be exceeded. After establishing equilibrium xenon conditions at 90 percent power, the repositioned rod groups were: Group I at 64 inches, Groups II and III at 66 inches, and Group IV at 57.3 inches. Before the repositioning of control rods, rod Groups I, II, and III were at 69 inches and the Group IV rod height was 52.4 inches.

The resulting perturbation in the axial flux and power distribution was apparently small, such that no detectable power oscillation as monitored by the core instruments was observed. The test was terminated after about 35 hours when the Group IV rods reached 58.9 inches.

The positions of the Group IV rods during the test are shown in Figure II-27. Data recorded by the seed fuel thermocouples for selected seed clusters are also illustrated in Figure II-27; seed cluster locations by coordinates are indicated in Figure II-2. During the period of instrument observation, no significant variation of the seed fuel temperatures was evident to indicate that an axial power oscillation developed, and no resulting rod position fluctuation similar to that observed on previous occasions was observable. Thus this test, which was performed when the control rods were controlling in the upper portion of the core, was inconclusive with regard to verifying an oscillatory xenon distribution as the cause of the control rod fluctuations.

Radial Flux Perturbation Test

A radial flux tilt test was performed during the reduced power and temperature operation of Seed 2 after the end of full power lifetime. In this test an asymmetric power distribution was caused by deliberate misalignment of selected rods in the controlling rod group, and the nuclear instruments and seed thermocouples were observed as an indication of the resulting power oscillation. The rods were subsequently realigned while observation of instrumentation continued.

In order to achieve a uniform xenon distribution prior to misalignment of control rods, the reactor was maintained at a constant 75 percent power level with an average coolant temperature of 475^{o}F until equilibrium xenon was attained. During this power run, rod Groups I through III were positioned at approximately 71 inches and control of the reactor was by means of the Group IV (subgroup 8) rods. Each of the four number 8 rods is identified with respect to rod number and core location in Figure II-2. In order to have the first increase in local power occur in the most fully instrumented core quadrant the number 81 and 83 Northwest - Southeast diagonal rods were selected to be misaligned equally above and below the average rod bank height of the two remaining aligned rods.

The radial flux tilt was initiated by misalignment of the number 81 rod 2.5 inches below and the number 83 rod 2.5 inches above the average rod bank height of the aligned number 82 and 84 rods. Following the initial rod misalignment, which was accomplished by temporarily transferring the individual rods 81 and 83 to a spare inverter, control of the reactor was maintained for the next 82 hours by moving the number 8 rods as an asymmetrical bank on a single inverter. The positions of the subgroup 8 rods are shown in Figure II-28, and the nuclear instrument channel and seed exit water thermocouple power ratios which correspond in time to the rod motions are shown.

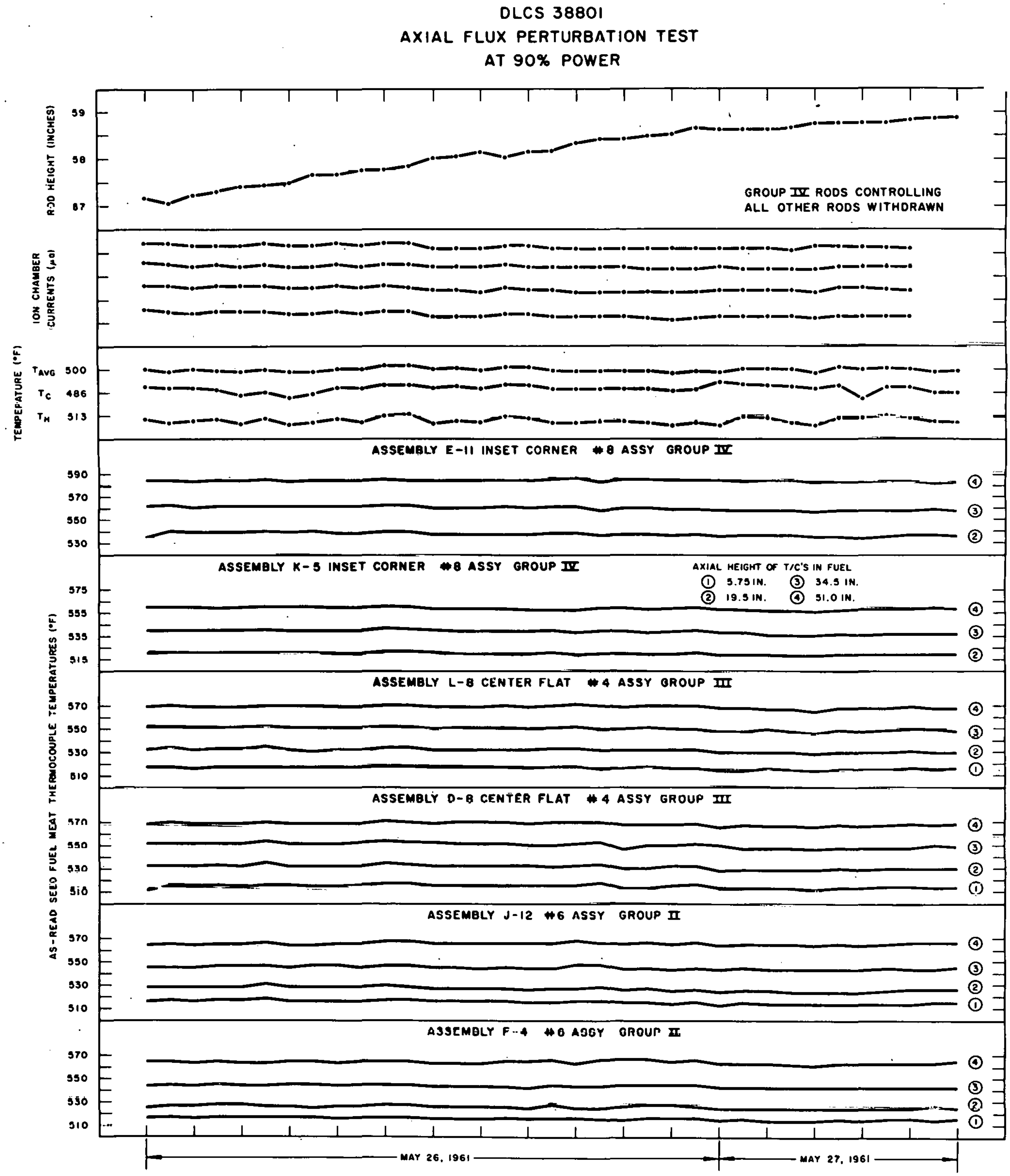

Figure II - 27. Seed 2 Reactor Instrument Readings vs Time.

The data from the nuclear instruments were processed to remove the effect of changing power levels and to remove the effect of different characteristic outputs from the four detectors. Each individual quadrant power ratio represents the relative amount of power fluctuation occurring in that quadrant from a nominally unperturbed power condition. As indicated by the nuclear instruments after the initial misalignment of the number 81 and 83 rods (Figure II-28 Part 2), a maximum power tilt of approximately ± 20 percent occurred in the B and C quadrants; after the realignment of rods (Figure II-28 Part 3), a maximum tilt of approximately ± 15 percent occurred. This difference in the magnitudes of the initial peak oscillations reflects the power asymmetry of ± 5 percent imposed by the asymmetric control rod configuration.

Seed exit water thermocouple data were utilized to provide a measure of the degree of power tilt to compare with the nuclear instruments. The thermocouple power ratio quantity denoted as power tilt by corners is defined as the numerical average of individual power tilts for the numbers 2, 6, and 8 seed cluster locations in each core quadrant, and the individual power tilts are defined as the ratio of the individual cluster coolant temperature rise, ΔT, to the average of the ΔT's for all symmetrically similar clusters. Thus, power tilt by corner should be a measure of the actual power tilt as indicated by coolant temperature rises in the various clusters.

During the first portion of the test when the controlling rods were misaligned (Figure II-28 Part 2), power tilts of up to nearly 30 percent are indicated by the thermocouples, in contrast to the 20 percent tilt indicated by the nuclear instruments (NIS). Several factors tend to indicate that the thermocouple tilt analysis is the more realistic measure of local power asymmetrics during rod misalignment. Since the seed exit water temperatures in certain clusters of the core quadrant, where power was greater than average, approached the limitations imposed on the basis of 100 percent power operation, the power in the high quadrant approached 100/75 times that in the average quadrant, corresponding to a maximum local power asymmetry of 33 percent. Comparison of the oscillatory behavior following the misalignment of the rods with that experienced upon re-aligning the rods indicates that a permanent local asymmetry component was introduced when the rods were misaligned, according to thermocouple data, whereas a less distinct tilt component is evident from the nuclear instrument data. A definite local component is expected on physical grounds (rod misalignment), and the thermocouples so indicate. The less pronounced indication of permanent asymmetry from the nuclear instruments can thus be interpreted as experimental evidence that these instruments portray a smeared image of the core quadrant, which does not reflect local power perturbations caused by rod misalignment to their full extent.

During the later portion of the test, when the rods were repositioned in normal, symmetrical alignment (Figure II-28 Part 3), power asymmetrics indicated by the nuclear instruments were comparable in magnitude to those determined by means of the core thermocouples. This agreement between thermocouple and NIS indications during oscillation is consistent with Seed 1 experience, which showed the same agreement among instrument during both spontaneous and experimentally induced oscillations. The present test thus provides additional assurance for the criteria utilized on Core I in establishing procedures to deal with spontaneous oscillations.

Power asymmetries as indicated by the nuclear instruments for the portion when the control rods were realigned (Part 3 of Figure II-28) are shown to be damped and to have a periodic behavior. The observed period for these oscillations is approximately 25.2 hours. In order to describe the damped behavior of the oscillations, the experimental data were fitted to the form:

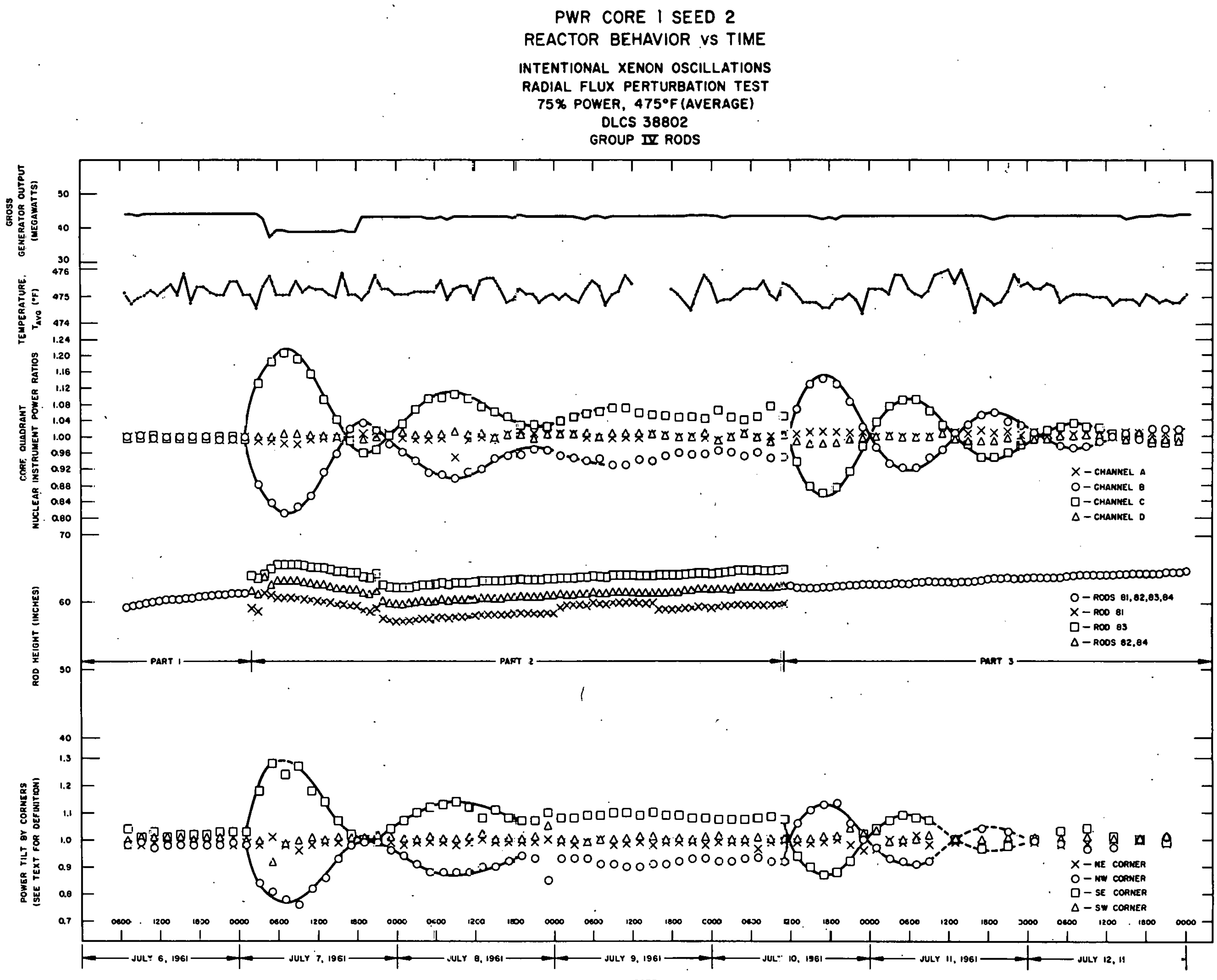

Figure II-28. Seed 2 Reactor Behavior vs Time.

$$y = e^{kt} \cos wt$$

From the data in this form, the damping factor k was found to be approximately -0.037 hr^{-1}. This corresponds to a reduction in the amplitude of the oscillations by a factor of two in 18.7 hours.

From Figure II-28 Part 3, it is noted that the oscillations in the radial flux perturbation test were strongly convergent. By comparision, slightly divergent oscillations were observed during an intentional xenon oscillation test performed at about 500 EFPH on Seed 1 with the Group II rods controlling, Reference 2. The convergent behavior of the Seed 2 radial oscillations indicates that the development of a spontaneous radial oscillation could not occur in Seed 2 unless an external perturbation (e.g. asymmetric rod motion) was to be applied at successive cycles in such a manner as to oppose natural damping of the oscillations. This fact is consistent with the observation that no detectable spontaneous oscillations developed in Seed 2 operation. The spontaneous oscillations which were encountered during Seed 1 operation are described in Reference 2.

Samarium Transient Measurements

At the conclusion of 75 percent reduced power operation, a samarium transient test was performed to determine the time dependent reactivity effect on the core from buildup of peak samarium concentration. The samarium transient test was conducted in a fashion similar to the xenon transient test (DLCS 15601) but with data collection in somewhat longer intervals. Before the performance of DLCS 15601 little experimental information on the samarium behavior of Core I was available. Data on the reactivity contribution from samarium as it affects the results of measurements of excess reactivity and temperature defect values have heretofore been obtained indirectly from other measurements or from calculations. The attendant uncertainties associated with samarium reactivity effects in turn affect the uncertainty in reactivity measurements obtained from other physics tests.

After shutdown from 75 percent power, the reactivity transient caused by the buildup and decay of xenon, and the buildup of samarium, was monitored by observing the critical positions of the controlling rod group and measuring reactor periods. The recorded time after shutdown and the reactivity values inferred from period measurements after analysis by the least squares technique yield the results given in Table II-J

Critical positions of the controlling Group IV rods, with all other rod groups withdrawn to approximately 71 inches, are shown as a function of time after shutdown in Figure II-29. The critical position of the Group IV rods was followed periodically for approximately 240 hours after shutdown to observe peak samarium conditions. The reactivity effect of xenon on the critical rod position is insignificant after approximately 100 hours after shutdown.

Group IV rod worths measured between 68 and 16 inches are illustrated in Figure II-30 as a function of rod height. By numerical integration under successive portions of the rod worth curve in Figure II-30, it was possible to construct the variation of the core reactivity with time after shutdown. Using the equilibrium xenon and samarium condition prior to shutdown as the reference point of reactivity change, the combined reactivity curve of xenon and samarium is shown in Figure II-31. The reversal of the trend of the reactivity curve in Figure II-31, at around 100 hours after shutdown, results from the decrease of the xenon reactivity effect followed by the increasing samarium effect after approximately 70 hours.

The elapsed time between shutdown and regaining of criticality was experimentally determined to be 30.9 hours. However, reactor moderator temperature at shutdown was 472°F while at the regaining of criticality it was 475°F. Adjusting the return time for the reactivity increase due to the negative temperature coefficient resulting from a 3°F temperature differential, the adjusted return time was 30.7 hours. Using this value for the return time in equation (1), the calculated average seed thermal flux level is given in Table II-J.

During Seed 1 operation, no samarium transient test was performed, although xenon transient tests were monitored for approximately 60 hours after shutdown. This interval is not adequate to observe the samarium poison effect. Results of the present samarium transient test show a measurable reactivity effect caused by the buildup of samarium, which previously had not been experimentally demonstrated for Core I.

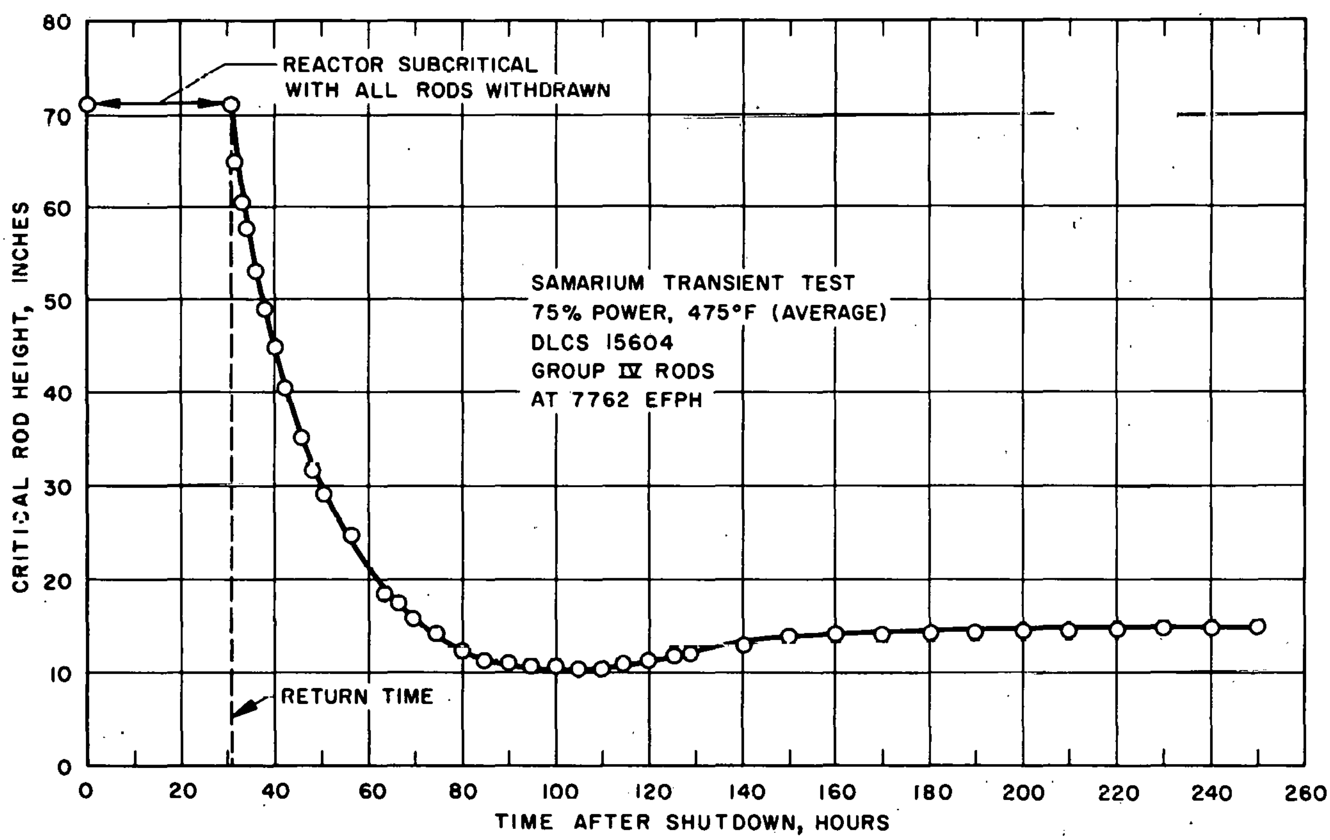

Figure II-29. Critical Rod Group Height vs Time after Shutdown.

TABLE II-J

SEED 2 XENON AND SAMARIUM TRANSIENT MEASUREMENTS

	Equilibrium Xenon Critical Rod Position	Minimum Critical Rod Position	Peak Samarium Critical Rod Position
Control Rod Group IV			
Critical Height inches	70.9	10.8	15.6
Time after shutdown (hrs)	0	105.	240.
Reactivity change following shutdown (% $\Delta\rho$)	0	2.49	2.42
Average seed thermal flux level, equation (1)			
:		5.7×10^{13} n/cm^2 sec	

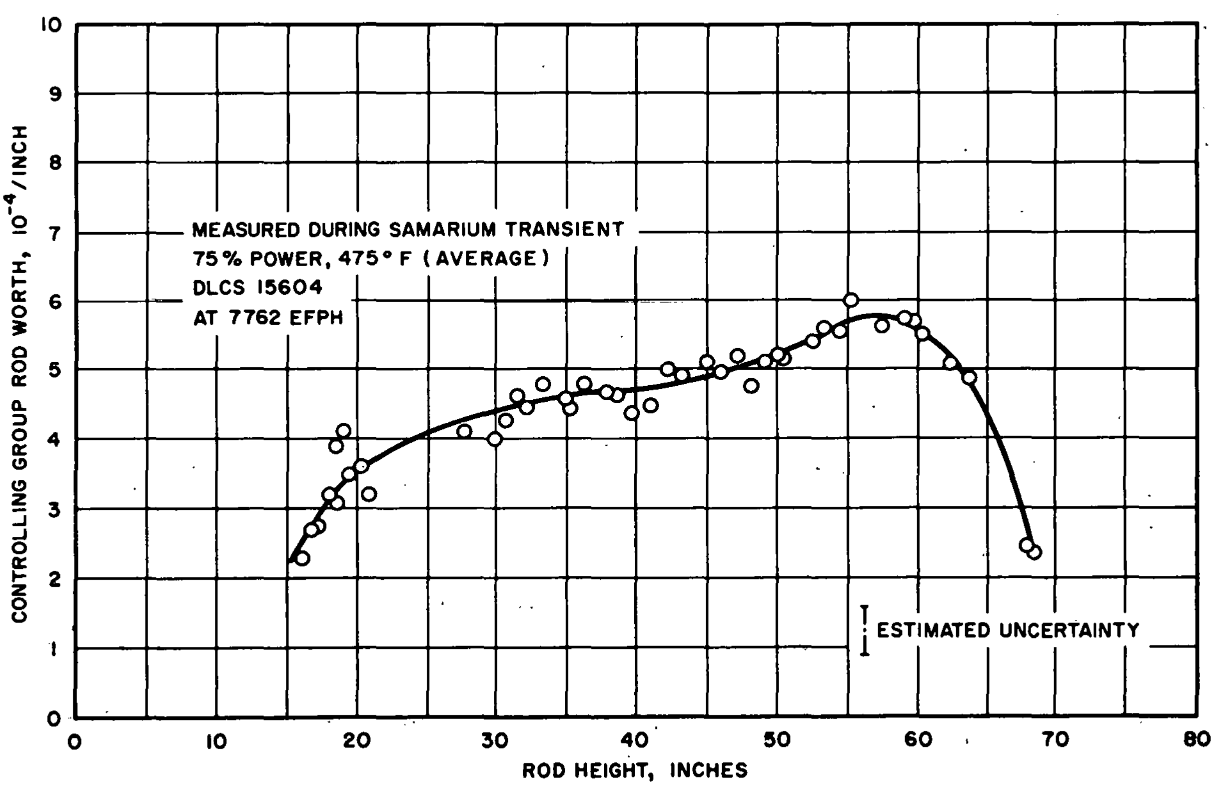

Figure II-30. Group IV Rod Worth vs Rod Height.

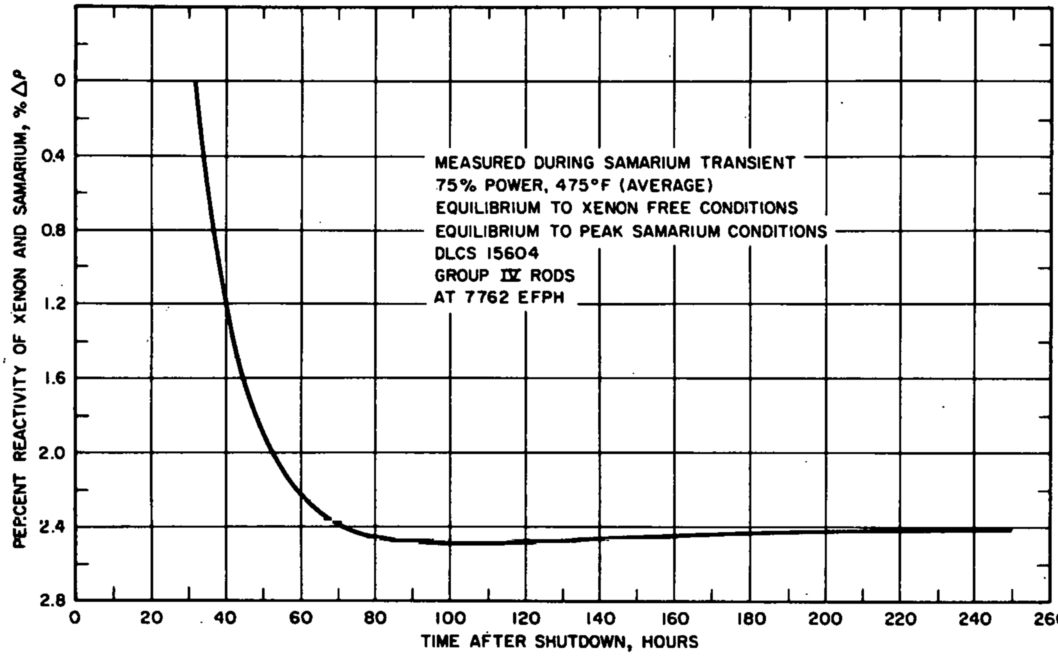

Figure II-31. Percent Reactivity of Xenon and Samarium vs Time after Shutdown.

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PART III

REACTOR THERMAL AND HYDRAULIC PERFORMANCE

CHAPTER 1

COMPARISON OF SEED 2 WITH SEED 1

Comparison of Fuel Elements

Seed 2, like Seed 1, consists of 32 clusters, each containing highly enriched uranium-zirconium alloy fuel in rectangular plates. For Seed 2 the total fuel loading was increased from 75 to 90 kg of U^{235} to provide increased seed lifetime. To compensate for the reactivity of the additional fuel, 170 grams of boron were added as a burnable poison. The number of fuel plates and coolant channels in each seed subassembly was increased from 15 and 16 in Seed 1 to 17 and 18, respectively, in Seed 2, thereby increasing the total heat transfer surface in the seed by about 15 percent. Seed 2 fuel plate and channel thicknesses were 0.066 inch in contrast to 0.069 inch in Seed 1. Of the 17 fuel plates in each Seed 2 subassembly, the three plates adjacent to each end contain the boron poison uniformly distributed in the alloy fuel, with the other 11 plates containing no boron. This arrangement of burnable poison was selected to reduce the peaking of the thermal neutron flux.

Except for 3 of the 113 blanket fuel assemblies, the blanket remained unchanged for Seed 2 operation. One blanket fuel assembly, F-2 (consisting of 7 bundles of fuel rods) was replaced by a new blanket fuel assembly because of an apparent fuel rod defect. Single rod bundles (consisting of 120 fuel rods) were removed from two other assemblies, E-6 and H-5, for detailed analysis of the irradiated fuel; these two bundles were replaced by new rod bundles.

Of the 32 hafnium control rods used in Seed 1, 31 were reused in Seed 2. One control rod was removed for evaluation and was replaced by a new, unirradiated control rod of the same design.

Table III-A compares the significant parameters for Seed 2 and Seed 1.

All fuel and control rods performed satisfactorily during Seed 2 lifetime. The additional 15 kg of U^{235} in combination with changes in blanket reactivity provides more than 2000 EFPH additional lifetime. The thermal and hydraulic performance of the core during Seed 2 operation is discussed further in Chapter III-2 and III-3.

A Comparison of the Instrumentation Designs of Core I Seed 1 and Seed 2

Core I instrumentation may be categorized as follows:

1. Seed metal and inlet water temperature instrumentation
2. Seed exit water temperature instrumentation
3. Blanket exit water temperature instrumentation
4. Auxiliary seed exit water temperature instrumentation
5. Flow measurement instrumentation
6. Failed element detection and location instrumentation
7. Blanket inlet water temperature instrumentation

TABLE III-A

NUCLEAR, THERMAL AND HYDRAULIC PARAMETERS

Property	Core I Seed 1	Core I Seed 2
Fuel plate thickness	0.069 in.	0.066 in.
Fuel alloy	U-Zircaloy-2	U-Boron Zircaloy-2 and U-Zircaloy-2
Fuel plate loading, per plate		
U^{235}	39.2 g	43.33 g std 37.89 g (with boron)
Boron	None	0.22 g
Number of fuel plates	1914 (total)	1408-2.1 in. fuel-width 768-2.0 in. fuel-width*
Total seedloading		
U^{235}	75 kg	90 kg
Boron (natural)	None	170 g
Actual core lifetime	5806 EFPH	7900 EFPH
Heat transfer area	3855 ft^2	4415 ft^2
Total cross section for flow	2.20 ft^2	2.379 ft^2
Coolant velocity (3-loop)	19.8 ft/sec	18.2 ft/sec
Nominal thermal power output (including blanket)	231 Mw	231 Mw
Design thermal power output (3-loop)	242.7 Mw	242.7 Mw
Primary coolant flow rate (3-loop operation)	23.06 x 10^6 lb/hr	23.4 x 10^6 lb/hr
Average primary coolant temperature	500°F	500°F
Average primary coolant pressure	1800 psia	1800 psia

* Reduced fuel width is required for the three fuel plates at each end of each subassembly to permit thicker side plates near the subassembly corners where the highest mechanical stresses occur.

Alterations in all but the last three instrument systems listed above were effected in conjunction with Seed 1 - Seed 2 refueling. Improvement in reliability resulted, as discussed later in this chapter. Locations of these instruments are shown in Figures III-1 and III-2.

Seed Metal and Inlet Water Temperature Instrumentation

Because of the numerous thermocouple sheathing failures experienced by the seed metal and inlet water thermocouples of the Seed 1 design, the sheath material of the Seed 2 thermocouples was AISI 347 stainless steel rather than the AISI 304 type. AISI 347 stainless steel is stabilized by the addition of columbium to prevent chromium carbide precipitation and resultant intergranular embrittlement, which is believed to have been a contributing factor in the development of cracks in the Seed 1 thermocouple sheaths. In addition, it was possible to increase the sheath thickness slightly by utilizing a 0.053 inch OD sheath rather than the previously used 0.040 inch size.

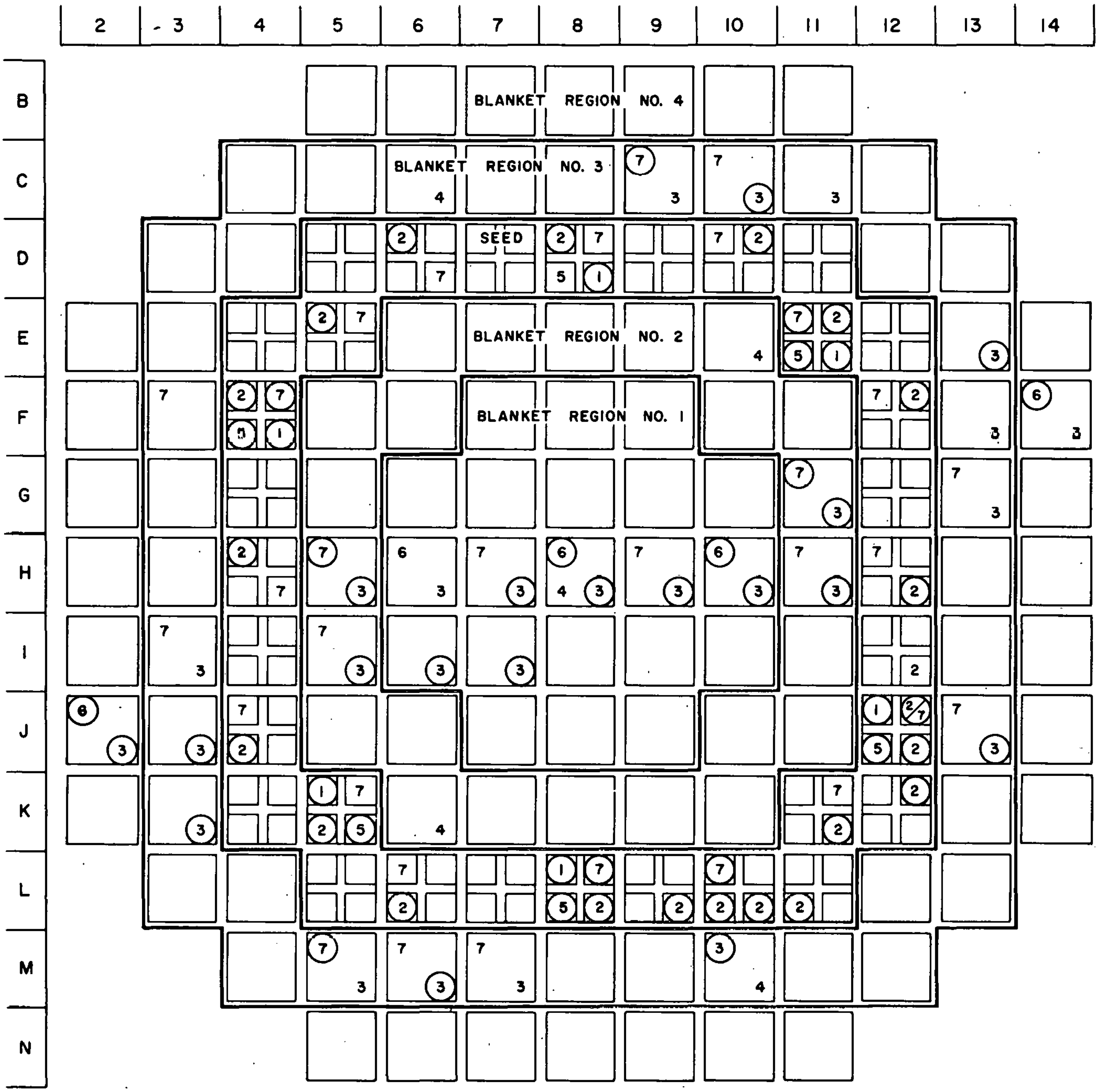
The major design change, however, was the provision for replacing individual thermocouples. In the Seed 1 design, the seed metal thermocouples were inserted into short horizontal wells between the two meat thicknesses of the instrumented fuel plate, led from there to the upper portion of the seed cluster where they were coiled on a drum which remained in the cluster as it was lowered into the reactor. Thereafter, the drum was lifted to the reactor head and the thermocouple leads were connected to the terminal posts of a terminal box mounted on the head. The instrumentation could be replaced only by installation of a complete spare seed cluster.

The Seed 2 design allowed each thermocouple to be individually positioned in the instrumented fuel plate by providing guide tubes leading from the terminal box on the vessel head to vertical thermocouple wells in the fuel plate. In order to effect thermocouple replacement, it was necessary only to cut the thermocouple closure weld on the reactor vessel head, withdraw the faulty lead, insert the replacement, and remake the weld. However, while the Seed 1 clusters contained seven thermocouple leads, the Seed 2 design permitted the installation of only five leads, one being the inlet water temperature thermocouple.

The intermediate thermocouple junction in the Seed 1 thermocouple terminal box was eliminated. The Seed 2 terminal box employed only one junction, that to the plug type connector. Furthermore, the junction was made and tested entirely in the shop, providing a more reliable initial installation which required less working time on the reactor head. Although the thermocouple wires were of very small diameter, as in the Seed 1 design, the junction was made mechanically rugged by potting it with a thermosetting resin.

The terminal box height above the reactor vessel head was increased to place the connector junction at a lower ambient temperature as compared to the Seed 1 installation, and, as a result, both the magnitude and dispersion of the calibration correction factors were reduced.

In addition, provision was made to subtract the voltage outputs of the Seed 2 inlet water thermocouples and exit water thermocouples to obtain a direct reading core temperature difference. However, a combination of equipment problems, unsatisfactory calibration, and failure of two inlet thermocouples prevented core temperature difference from being satisfactorily measured in this manner.



INSTRUMENT LEGEND

1. SEED FUEL PLATE THERMOCOUPLE
2. SEED EXIT WATER THERMOCOUPLE
3. BLANKET EXIT WATER THERMOCOUPLE
4. BLANKET INLET WATER THERMOCOUPLE
5. SEED INLET WATER THERMOCOUPLE
6. FLOW NOZZLE
7. MODIFIED VENTURI

○ - OPERABLE THERMOCOUPLE OR CALIBRATED FLOW D/P CELL

NO CIRCLE - INOPERABLE THERMOCOUPLE OR UNCALIBRATED FLOW D/P CELL

EARLY IN SEED 2 LIFE JULY 1, 1960

Figure III-1. PWR Core I Seed 2 Instrumentation.

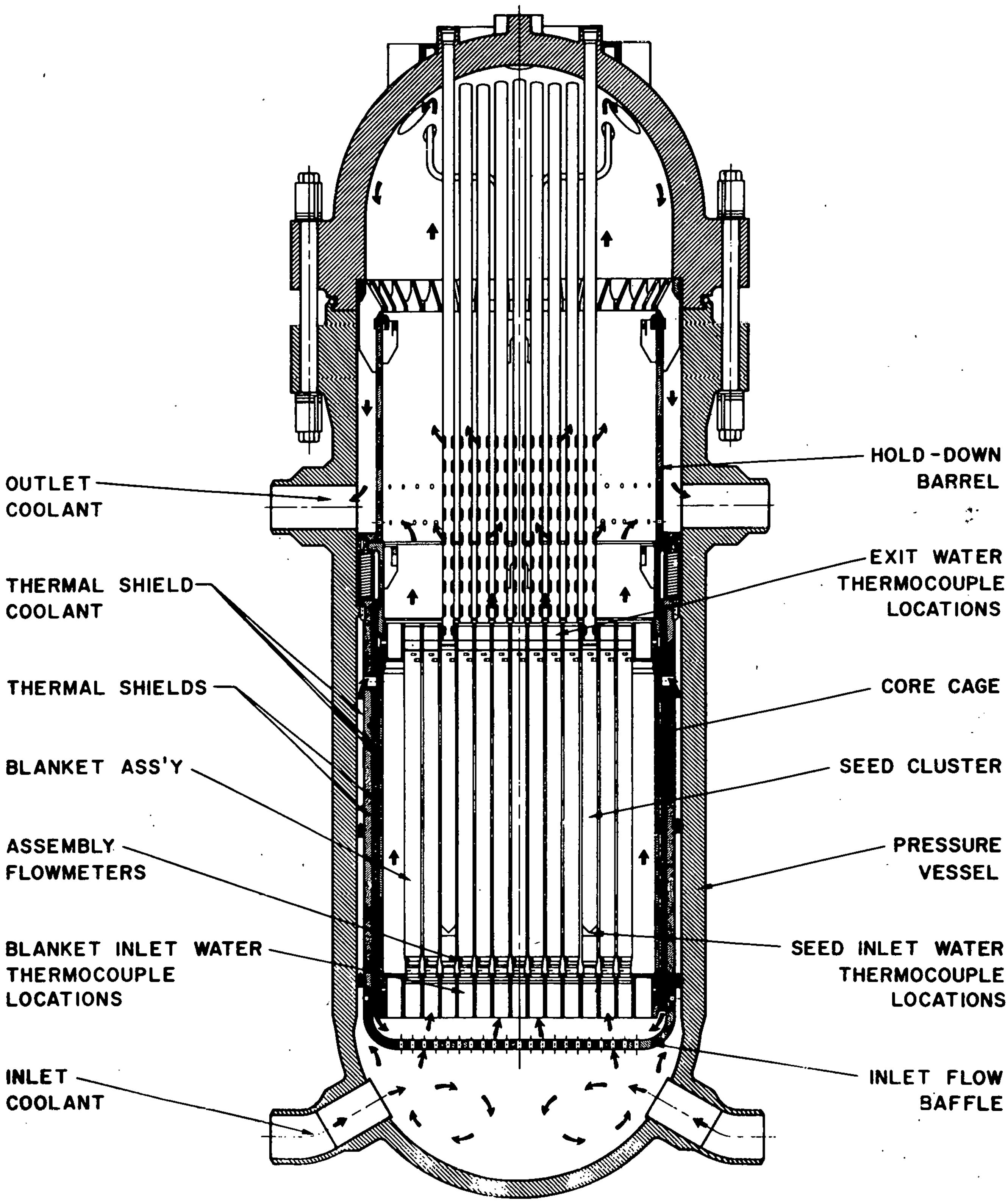


Figure III-2. PWR Core I Reactor Flow Paths and Core Instrumentation Locations.

Seed Exit Water Temperature Instrumentation

The same improvements in the thermocouple sheathing material and terminal box designs described above were incorporated in the Seed 2 exit water instrumentation design. The number of leads, their diameter, their core location, and the manner in which they are installed remained the same as in Seed 1. However, all the thermocouples had non-grounded junctions to allow the subtracting of inlet and exit water thermocouple outputs as previously described.

Blanket and Auxiliary Seed Exit Water Temperature Instrumentation

One blanket exit water instrumentation (BEWI) assembly (H-9) and both auxiliary seed exit water instrumentation (ASEWI) assemblies in Seed 2 incorporated the replaceable thermocouples previously described. These three assemblies are also equipped with the improved thermocouple sheathing material and terminal box designs of the seed metal system. The number of leads, their diameter, and their core locations remained the same as in Seed 1. However, two leads in each of the ASEWI assemblies contained non-grounded thermal junctions (I-12, K-12, L-11, L-9) to permit subtracting their outputs from those of the seed inlet water temperature thermocouples at core location J-12.

Performance of Core Thermocouples During Seed 2 Operation

The replacement instrumentation of new design installed for Seed 2 operation proved very successful and represented a large improvement over the earlier design used for Seed 1. Of the 77 thermocouples installed during Seed 1 - Seed 2 refueling, 75 were operating at Seed 2 start-up and 71 were still giving useful information at the end of Seed 2 life. The calibration of these new thermocouples was more stable and the corrections were smaller than those of the earlier design, probably because of improvements in the junction box location and design. The replacement features of these thermocouples were not utilized during Seed 2 operation because failures were relatively few and occurred for the most part after sufficient information had been obtained to provide assurance that the temperatures at the locations of the failed thermocouples could be inferred from symmetrically located instruments which were still functioning. Discussion of the information obtained from the core thermocouples during Seed 2 operation is contained in Chapters III-2 and III-3.

CHAPTER 2

HYDRAULIC PERFORMANCE

Introduction

Testing performed during Seed 2 operations has essentially substantiated the major phases of reactor hydraulic design:

1. Predicted core flow distribution to fuel assemblies in the five core flow regions (Figure III-1) was confirmed prior to power operations.

2. A continuing test of reactor pressure drop variations with time has established that there are no consistent trends in this parameter and only small variations with time.

3. Test data were obtained which validate the design data based on airflow studies of mixing in the lower plenum chamber.

4. Finally, characteristic reactor flow coastdown curves have been determined over a five-second interval after initiation of the loss of flow.

Each of the above phases is discussed more fully below with particular attention to the agreement of design and measured values.

Core I is a single-pass core. Coolant enters the lower plenum chamber through four inlet nozzles at the bottom of the reactor vessel, as shown in Figure III-2. After mixing in the lower plenum chamber, flow is apportioned along parallel paths to the core inlet through either the flow baffle (85 percent) or the thermal shields (15 percent). The coolant then moves upward through the core and empties into the upper plenum chamber where it mixes. The main portion of the flow is then directed upward inside the holddown barrel and then downward between the barrel and the reactor vessel wall to the four outlet nozzles.

Core Flow Distribution

The design distribution of coolant in the core is listed below.

Seed (32 assemblies)		37.2%
Fuel region	34.9%	
Control rod channels	2.3%	
Blanket region 1 (21 assemblies)		7.4%
Blanket region 2 (24 assemblies)		17.5%
Blanket region 3 (40 assemblies)		27.1%
Blanket region 4 (28 assemblies)		7.3%
Labyrinth seal leakage		2.0%
Bottom plate flow meter adapters		1.0%
Instrument tubing		0.5%
		100.0%

The total reactor coolant flow rate is determined by intersection of the reactor pressure drop characteristic curve with the available reactor pressure head curve, which is constructed by subtracting coolant loop pressure losses from the main coolant pump head at several different loop flow rates.

Figure III-3 shows the intersection of the available reactor pressure head curve and the reactor pressure drop characteristic curve. At this point both flows and pressure drops are balanced and the reactor flow is therefore determined.

To make the maximum use of coolant, it was necessary that flow be distributed among the seed and four blanket regions in proportion to the region heat generation rates, maintaining approximately equal outlet temperatures for all fuel assemblies. To accomplish this, inlet orificing was required in three of the four blanket regions. The blanket orificing arrangement selected for Seed 1 operation was also satisfactory for Seed 2 performance and, therefore, was not altered. Table III-B summarizes some pertinent core flow parameters for three-loop flow conditions.

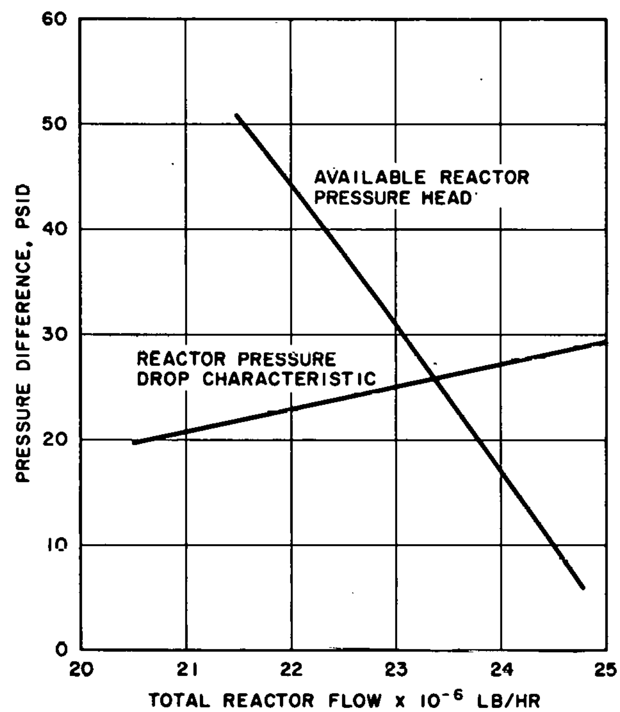


Figure III-3. Hydraulic Balance Point for Three Loop Operation.

One of the first tests (DLCS 18201) performed subsequent to Seed 2 installation was the determination of flow distribution within the core. A comparison of predicted and measured flow rates is presented in Table III-C. Predicted nominal readings were 100 percent on four loops and 79 percent on three loops. However, the values in Table III-C were adjusted to account for variable pump flows since A, B, and C coolant loops contained similiar pumps while D loop contained a pump which has a different head-flow characteristic.

TABLE III-B

CORE FLOW PARAMETERS

Parameter	Blanket Region I	II	III	IV	Seed
Three-loop flow rates, lb/hr-assembly	8.27×10^4	17.07×10^4	15.86×10^4	6.09×10^4	25.47×10^4
Subassembly ΔP, psi	5.8	24.8	21.4	3.2	24.9
Flowmeter ΔP, psi	3.3	0.5	0.4	1.8	1.0
Orifice ΔP, psi	16.8	*	4.1	20.9	--
Orifice plates, no./assembly	3	0	1	3	--
Orifice coolant velocity, ft/sec	24.6	--	35.6	27.4	--

* No orifice plate was required in Region II since only 0.6 psi difference existed between Region II and the seed.

TABLE III-C

COMPARISON OF PREDICTED AND MEASURED FLOW FOR THREE AND FOUR-LOOP OPERATION

Core Region	Flowmeter Type	Loops in Operation and Average Flowmeter Readings (% of design value): ABCD Pred.	ABCD Meas.	ABD Pred.	ABD Meas.	BCD Pred.	BCD Meas.	ABC Pred.	ABC Meas.	Prediction Minus Measurements Average (%)
Seed	Venturi	98.2	100.2	76.7	76.6	76.7	76.1	79	78.5	+0.2
Blanket I	Nozzle	98.2	87.6	76.7	69.8	76.7	69.2	79	71.7	-8.1
Blanket II	Venturi	98.2	98.9	76.7	77.5	76.7	75.5	79	79.9	+0.3
Blanket III	Venturi	98.2	95.5	76.7	76.4	76.7	74.8	79	76.5	-1.8
Blanket IV	Nozzle	98.2	98.6	76.7	77.8	76.7	77.0	79	79.1	0.5

The seed and blanket regions 2 and 4 show very good agreement (within 1 percent) between average predicted and measured flowrates. Blanket region 3 apparently has a small flow deficiency (-1.8 percent) but this is within the accuracy of the instrumentation (±2 percent). Blanket region 1 has a measured flow 8.1 percent less than predicted on the average, but this is based on data from assemblies with nozzle-type flowmeters which have had a history of erratic operation. The two venturi-type flowmeters in this region are inoperable. During Seed 1 operation the blanket region 1 venturis indicated approximately design flow, while the nozzles indicated flowrates 6 to 10 percent below design. During Seed 2, blanket region 4 assembly J-2 nozzle flowmeter has consistently indicated flows about 5.5 percent above design. Because of this conflicting information and the history of erratic operation, no conclusions can be made based on the indications of the flow nozzles.

It is therefore concluded that Seed 2 core region flowrates are essentially as designed within flowmeter instrumentation accuracies. This is necessarily based on Seed 1 as well as Seed 2 results because of the erratic operation of the nozzle type flowmeters in blanket regions 1 and 4.

Distribution of flow among fuel clusters in a quarter-scale flow model of the PWR reactor has been studied using airflow to simulate coolant flow (Reference 1). The maximum variation in flow to any core assembly was ±2.1 percent of predicted assembly flow for three-loop flow and ±1.5 percent for four-loop flow, based on the airflow data.

Reactor Pressure Drop

Predictions of reactor pressure drop were made utilizing seed and blanket assembly pressure drop characteristics, in conjunction with assumptions regarding other reactor pressure losses. Inlet nozzle losses were assumed to be one velocity head, and outlet nozzle losses were assumed to be half a velocity head. Holddown barrel losses were treated in a manner similar to the nozzles, and the flow baffle losses were determined from Battelle airflow model testing. The predicted reactor pressure drops for three and four-loop flow are:

	3 Loops	4 Loops
Reactor flow x 10^{-6} lb/hr	23.4	29.5
Inlet nozzles ΔP, psi	8.4	7.5
Baffle ΔP, psi	2.0	2.0
Core ΔP, psi	25.9	40.9
Holdown barrel ΔP, psi	0.8	1.3
Outlet-nozzles ΔP, psi	3.4	3.0
Total reactor ΔP, psi	40.5	54.7

Measured pre-operational data (DLCS 1470105) indicate that the reactor pressure drops are 36.9 and 49.0 psi for three and four loops, respectively. These values are about 10 percent below predictions and are more than likely the result of minor conservatisms in the assumptions concerning inlet and outlet nozzle and holddown barrel losses, since reactor coolant flows are very close to design values.

Continuing investigation of reactor pressure drop with time (DLCS 35801) has indicated there are no detectable trends within the accuracies of the test instrumentation. As-read reactor pressure drop and flow variations with time are presented in Figure III-4. It can be seen that the majority of the pressure drop variations are accompanied by similar flow variations. In addition, a 1500-hour test without purification flow from June 23 to September 13, 1960 also showed no detectable change in reactor pressure drop characteristics. These results are apparently caused by the inherently low crud levels during steady state operations, usually from 1 to 4 ppb.

Inlet Plenum Mixing

The question of coolant mixing in the lower plenum of the reactor vessel arises when the inlet temperature to one of the non-instrumented clusters is required and only coolant temperatures at the inlet nozzles to the reactor are available. Quantitative information is necessary in order to evaluate the procedure by which a non-operating cold loop is returned to service in order to avoid cold water accidents. The calculations to determine the fraction of total core power generated in various core assemblies also depend on this information.

Mixing in the lower plenum of a quarter-scale airflow model of PWR was studied using SO_2 as a gaseous tracer in air as reported in BMI 1172 (Reference 2). Actual plant data were obtained during the performance of DLCS 1820201 in January 1958, during Seed 1 operation. The general spread in the data, however, was too great to either substantiate or invalidate the results of BMI-1172. At best, both the calculated and predicted values showed the same general trend in mixing in the few assemblies for which data were available. The test was performed again during Seed 2 lifetime. Typical results of the second performance of this test are shown in Figure III-5.

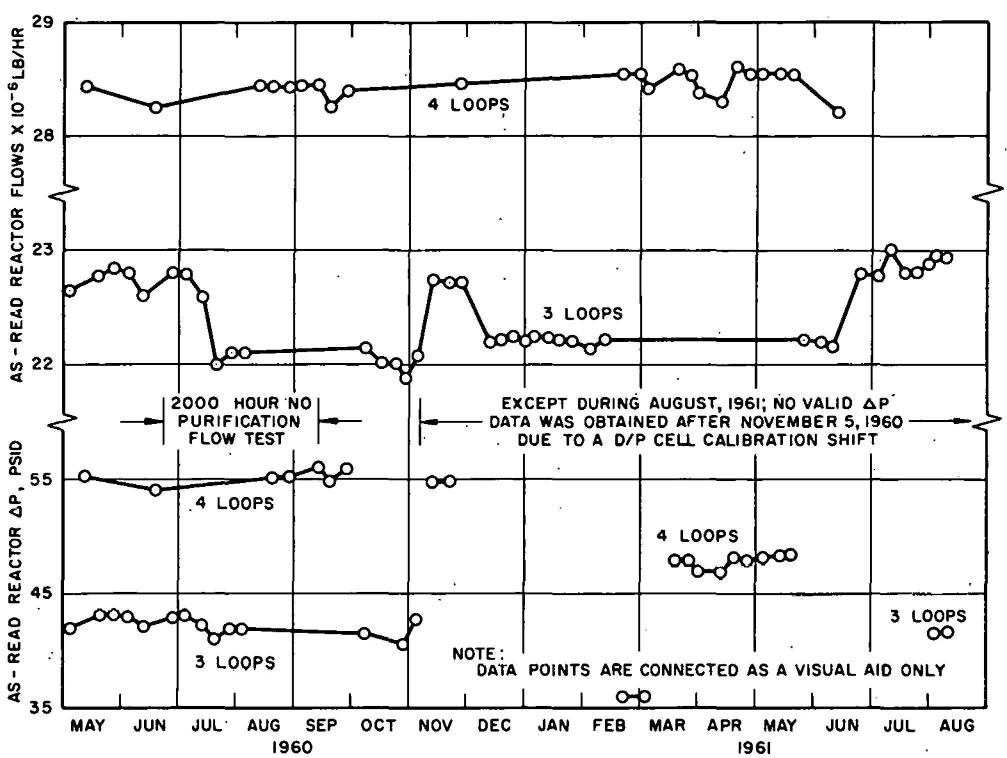


Figure III-4. Time Variation of Reactor Pressure Drop and Flow.

Close agreement was found between measured and predicted data, thereby substantiating the extent of mixing across the core as predicted by the Batelle Memorial Institute tests. For the majority of assemblies analyzed, the data were reproducible and within 5 percent of one another.

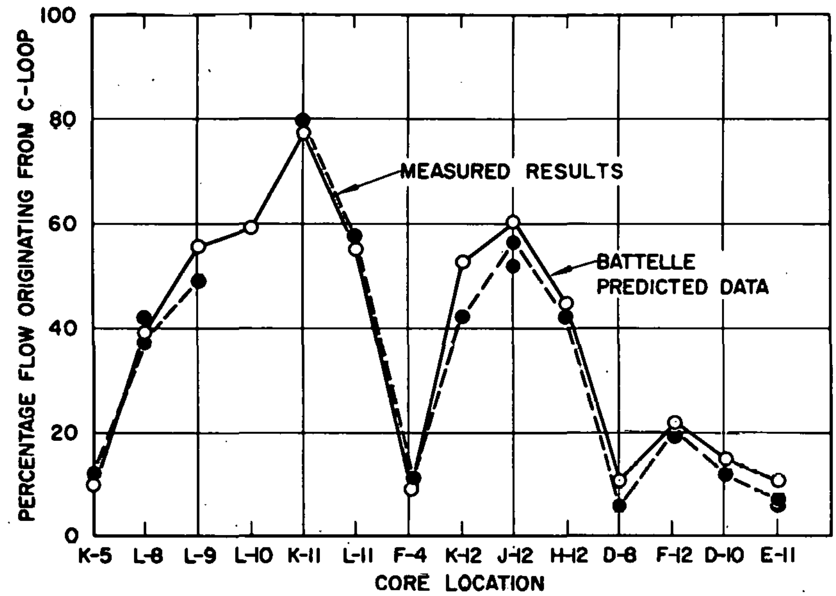


Figure III-5. Inlet Plenum Mixing Data-Measured and Predicted.

Another operational verification of the airflow model inlet plenum mixing was obtained during the test to determine the effect of steam generator performance on core power distribution (DLCS 3770101). For a portion of this test the inlet water temperatures for two of the coolant loops were 22.5°F higher than that of the other two loops. At this time inlet water thermocouples checked the airflow mixing temperatures within an average of 0.8°F.

Loss of Coolant Flow

In the study of core capability for Seed 2, it was found that near the end of Seed 2 lifetime, core capability was established by the blanket minimum departure from nucleate boiling (DNB) ratio at about 3 to 5 seconds after the start of a complete loss of flow accident (CLOFA). Data obtained during the loss of flow testing with Seed 1 were inadequate beyond approximately 2 seconds after the start of the transient. In Seed 2 tests, therefore, most emphasis was placed on better accuracy of flow coastdown measurements several seconds after pump shutdown.

Comparisons of design and measured Seed 2 curves are presented in Figures III-6 and III-7 CLOFA's with initial flow provided by four loops (4-0 CLOFA) and three loops (3-0 CLOFA), respectively. The CLOFA design flow coastdown curves for Core I have been revised to be consistent with results of Seed 2 testing. These are unchanged from the former design curves (determined from analysis and Seed 1 test results) in the first 2 or 3 seconds of coastdown, but beyond that time the CLOFA 4-0 curve is lower (by as much as 45 percent) than formerly, and the CLOFA 3-0 curve is higher (by as much as 20 percent) than formerly. The average Seed 2 corrected data are approximately to 10 to 20 percent above the revised design curves. Measured flow rates during the first second are considerably higher than design, but will not be used without a more detailed investigation to resolve such differences.

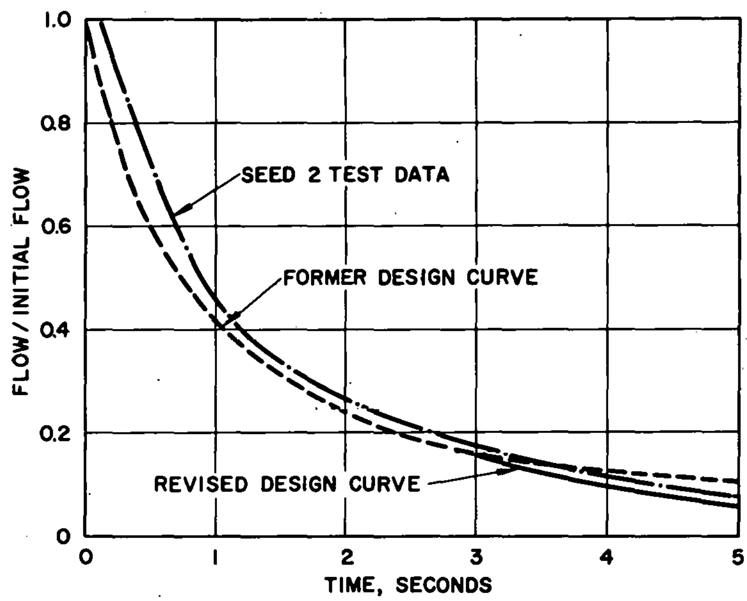


Figure III-6. CLOFA 4-0 Data.

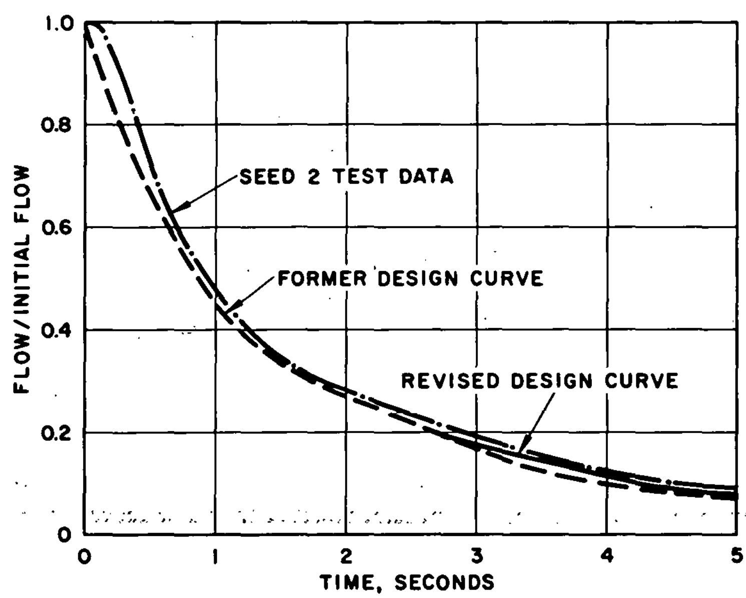


Figure III-7. CLOFA 3-0 Data.

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2. Studies of Mixing in the Lower Plenum of a Quarter-Scale Flow Model of the PWR Reactor, Battelle Memorial Institute, BMI-1172, February 22, 1957.

CHAPTER 3

THERMAL PERFORMANCE

Introduction

During Seed 2 operation the thermal performance of the reactor was monitored with the aid of instrumentation in the core, the primary loop, and the secondary loop. Core thermal instrumentation was used primarily for two purposes: (1) to check the adequacy of the theoretical model used for the calculation of reactor power capability, and (2) to provide a means of measuring phenomona such as power tilt and oscillations which, at the present state of the art, do not lend themselves readily to theoretical evaluation. Only during special testing was core thermal instrumentation used as an operational guide. The thermal and hydraulic instrumentation in the primary and secondary loops was used to furnish supplementary thermal performance information in addition to its main function of plant control. Measurements obtained from the core instrumentation during Seed 2 operation indicated that core thermal performance was satisfactory.

Core Thermocouple Predicted and Measured Temperatures at the Beginning of Seed 2 Life

Summary

At the beginning of Seed 2 life a comparison was made of core thermocouple data with calculated temperatures. The thermocouple data were selected from power operations at equilibrium xenon where the controlling rod group height was within ± 1 inch of the height for which the nuclear design data were available. Thus a comparison was possible in which a major parameter, rod height, was matched well with the rod height assumed in the calculations of predicted temperature.

A conclusion which can be made on the basis of the comparison is that calculated temperatures and measured temperatures agree satisfactorily. Table III-D shows the degree of agreement obtained.

TABLE III-D

AGREEMENT BETWEEN MEASURED AND CALCULATED TEMPERATURE VALUES

	Number of Readings Within X% of Calculated Values	
Temperature	X = 10	X = 15
Seed Assembly Coolant ΔT	26 out of 33	30 out of 33
Blanket Assembly Coolant ΔT	11 out of 18	15 out of 18

Operating Conditions When Data Were Taken

On August 21, 1960, at 0330 hours the conditions were:

Group II at 20.03 inches, gross output - 66 Mw, nuclear instruments - 105% (242.6 Mw thermal), 4 loops (fast speed), total flow - 29.2×10^6 lb/hr, Seed 2 EFPH - 1700, inlet temperature - 488°F.

On August 28, 1960, at 2300 hours the conditions were as above except for:

Group II at 21.76 inches, Seed 2 EFPH - 1880.

Assumed Operating Conditions for Calculated Temperatures

Group II at 21 inches, reactor power-231Mw, inlet temperature - 488°F, design assembly flowrates, complete assembly coolant mixing, DRACO local power generation, 95% reactor power generated in core.

Fuel plate centerline temperatures were calculated for nominal and certain hot channel conditions. In computations for nominal fuel plate temperatures the following conditions were assumed:

Group II at 21 inches, 4 loops (fast speed), total flow - 29.52×10^6 lb/hr, reactor power - 231 Mw, reactor average temperature - 500°F, inlet temperature - 488°F, 95% reactor power generated in core, 2% core power generated in water, no lateral mixing of coolant, design region flowrates, average region pressure drop - channel pressure drop, DRACO local power generation with DRACO error correction factor and local to average correction factor, mechanical hot channel factors - 1.0, no control band or instrument uncertainties.

In the calculations of the upper limit temperature conventional mechanical hot channel factors were applied and the nuclear factors (relative power generations) were increased by 10 percent in all clusters except those with control rods fully inserted.

Coolant Temperature Rise Through Core Assemblies

Both the measured and calculated values of coolant temperature rise were obtained for a nominal power level of 105 percent of 231 Mw core thermal putput. The measured temperature rises were corrected using thermocouple calibration factors obtained July 31, 1960.

Seed Assemblies

A comparison of measured and calculated values indicated that the measured temperature rises were less than the calculated values for every assembly in which the instrumentation was operative. In fact only 7 of 33 temperature rises were within 10 percent of the calculated values. Those differences illustrated some problems associated with the prediction of local power distributions. As a result of power gradients within an assembly, coolant streams leaving the passages between fuel elements may have significantly differing temperatures. Mixing of these different streams of coolant occurs to some undetermined extent in the two feet of flow path in the upper extension bracket before the coolant reaches the single thermocouple by which the assembly exit water temperature is read.

While the hydraulic and thermal mixing cannot be analytically determined, estimates of the mixing effect have been made. They are based on temperature readings of an assembly instrumented specifically with three exit water thermocouples, on nuclear design cell calculations, and on the actual location of the exit water thermocouple with respect to the assembly. When these estimates of the mixing effect (mixing factors) were applied to the measured-calculated ratios, it was found that agreement of measured and calculated temperatures was quite good, with 26 out of 33 of the ratios falling within ± 10 percent of 100 percent and 30 of the ratios lying within ± 15 percent of 100 percent. Such agreement is considered to be quite adequate in the light of the uncertainties connected with the measurement of reactor power, temperature and flow, the coolant mixing, and thermocouple calibration, as well as uncertainties in nuclear design predictions of core power distribution. An example of the latter is the omission from the predicted values of a bias in the nuclear design calculations thereby causing power predictions to be high in assemblies containing control rods and low in assemblies without rods. The maximum value of this bias is 2°F in the temperature rise across an assembly at full power. Although no allowance was made for this difference in predicted temperature rises, it was considered in establishing core capability. Table III-E lists the measured and calculated temperature rises in the seed and the measured-calculated ratios.

Blanket Assemblies

Measured and calculated coolant temperature rises and measured-calculated ratios are listed in Table III-F. Thermocouple calibration corrections were applied to the measured temperature rises. A comparison of the blanket measured-calculated ratios indicates that in 11 out of 18 assemblies the ratios are within ± 10 percent of 100 percent and 15 out of 18 are within ± 12 percent of 100 percent. Since there are no control rods in the blanket, the power gradient across blanket assemblies is less severe than across seed assemblies. For this reason, and because blanket cluster tube sheets tend to enhance turbulent flow, no mixing factors were applied in the blanket. Thermal and hydraulic mixing in the blanket assembly in and above the fuel will nevertheless affect thermocouple readings. It is felt that the agreement of measured with calculated ΔT's in the blanket is quite satisfactory when the possible uncertainties enumerated above are considered.

TABLE III-E

SEED ASSEMBLY COOLANT TEMPERATURE RISE

Locations	Measured ΔT ($^{\circ}$F)	Calculated ΔT ($^{\circ}$F)	$\frac{\text{Measured } \Delta T}{\text{Calculated } \Delta T}$ Without Mixing Factor (%)	$\frac{\text{Measured } \Delta T}{\text{Calculated } \Delta T}$ With Mixing Factor (%)
J-4	27.5	31.22	88.0	106.0
H-4	19.9	26.37	75.5	90.9
D-8	16.1	26.37	61.0	73.4
D-10	28.6	31.22	91.6	110.3
E-11	26.0	30.26	85.9	103.4

TABLE III-E (Cont'd)

SEED ASSEMBLY COOLANT TEMPERATURE RISE

Locations	Measured ΔT (°F)	Calculated ΔT (°F)	Measured ΔT / Calculated ΔT Without Mixing Factor (%)	Measured ΔT / Calculated ΔT With Mixing Factor (%)
F-12	25.8	31.22	82.6	99.5
H-12	20.1	26.37	76.2	91.7
J-12	26.1	31.22	83.6	100.7
K-11	25.1	30.26	82.9	99.8
L-10	25.8	31.22	82.6	99.5
L-8	15.9	26.37	60.3	72.6
F-4	26.2	31.22	83.9	101.0
E-5	25.6	30.26	84.6	101.9
D-6	28.6	31.22	91.6	110.3
L-6	27.2	31.22	87.1	104.9
K-5	25.8	30.26	85.3	102.7
D-10	29.1	31.22	93.2	112.2
E-11	26.0	30.26	85.9	103.4
F-12	26.6	31.22	85.2	102.6
H-12	19.7	26.37	74.7	89.9
J-12	26.2	31.22	83.9	101.0
K-12	30.8	31.88	96.6	103.4
K-11	25.1	30.26	82.9	99.8
L-11	31.0	31.88	97.2	104.0
L-10	25.5	31.22	81.7	98.0
L-9	30.7	30.53	100.6	104.0
L-8	16.1	26.37	61.1	73.6
L-6	26.5	31.22	84.9	102.6
K-5	25.5	30.26	84.3	101.5
J-4	27.3	31.22	87.4	105.2
F-4	25.8	31.22	82.6	99.5
E-5	26.3	30.26	86.9	104.6
D-6	28.3	31.22	90.6	109.1

Note: Inoperative recorder points have been ommitted.

TABLE III-F

BLANKET ASSEMBLY COOLANT TEMPERATURE RISE

Location	Measured ΔT (^oF)	Calculated ΔT (^oF)	Measured ΔT / Calculated ΔT (%)
J-13	19.5	18.64	104.6
M-10	22.4	18.64	120.2
J-3	21.7	18.64	116.4
K-3	19.0	17.63	107.8
J-2	23.8	24.02	99.1
I-7	23.5	24.66	95.3
H-7	20.0	22.73	88.0
I-6	28.7	31.07	92.4
I-5	21.7	20.84	104.1
G-11	20.4	20.84	97.9
H-9	17.4	22.73	76.6
H-11	17.7	18.96	93.4
H-10	27.2	29.02	93.7
H-8	19.4	20.81	93.2
H-5	20.9	18.96	110.2
D-10	20.0	18.64	107.3
E-13	18.7	17.63	106.1

Note: Inoperative recorder points have been ommitted

Seed Fuel Plate Temperatures

Perhaps the most interesting instrumentation in Core I during Seed 2 life was the seed fuel plate temperature instrumentation.

Prior to Seed 2 startup, calculations were made for each of the seed fuel plate thermocouples. These calculations differed from the coolant temperature rise calculations previously described in that an attempt was made to describe, as closely as possible the power density at the termocouple location. Calculations were based on nominal channel dimensions to provide a lower limit for the prediction, and also on hot channel dimensions to provide an upper limit case. In these studies the effect of differences in flow from channel to channel was included by assuming that the design flow-rate applies to the average channel in the seed assembly containing the instrumented plate. The pressure drop across this average channel was then imposed across the instrumented plate channel using the local fluid properties to obtain flow past the instrumented plate.

In Figures III-8 and III-9 are plotted the predictions for four-loop operation at 104 percent power with the Group II control rods at 21 inches. Also indicated are the measurements from the two sets of data under the operating conditions described above. In all cases the measured values lie between

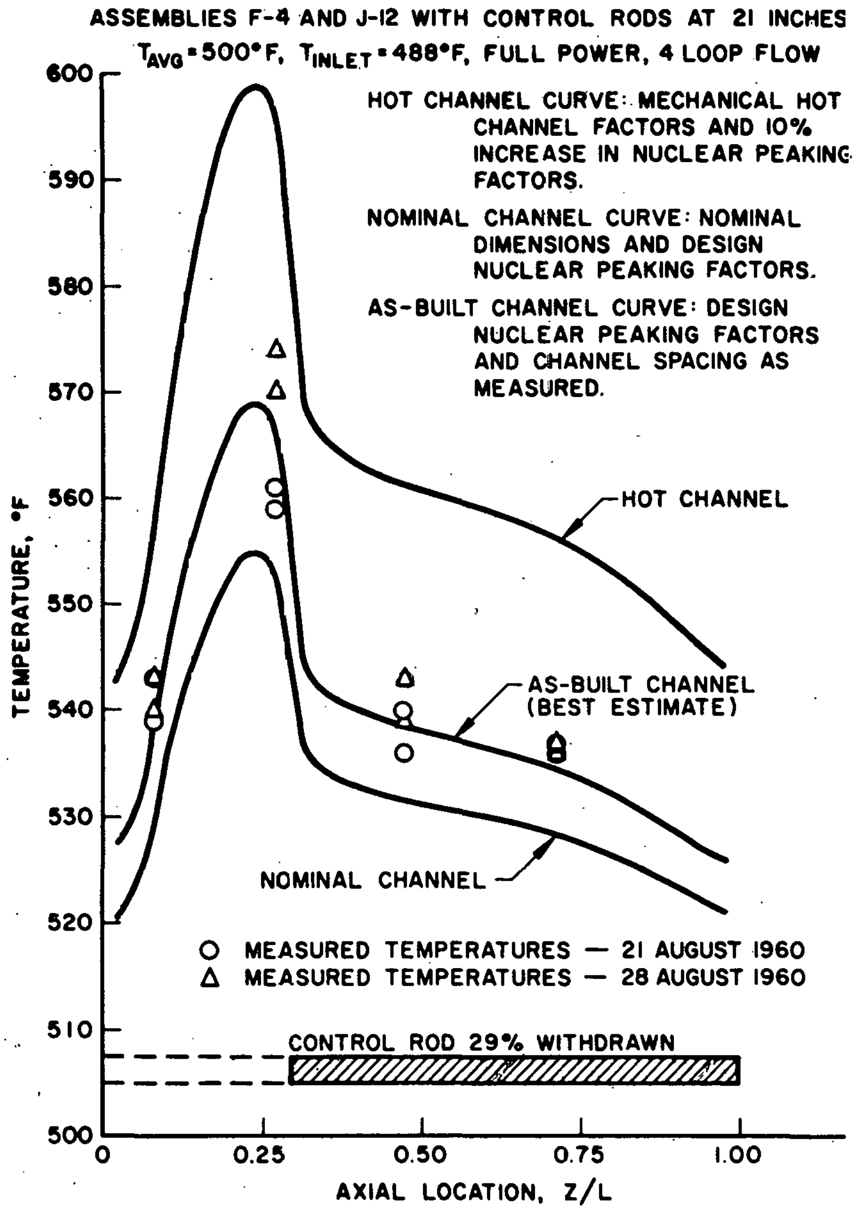


Figure III-8. PWR Seed Fuel Plate Temperature Predictions and Readings.

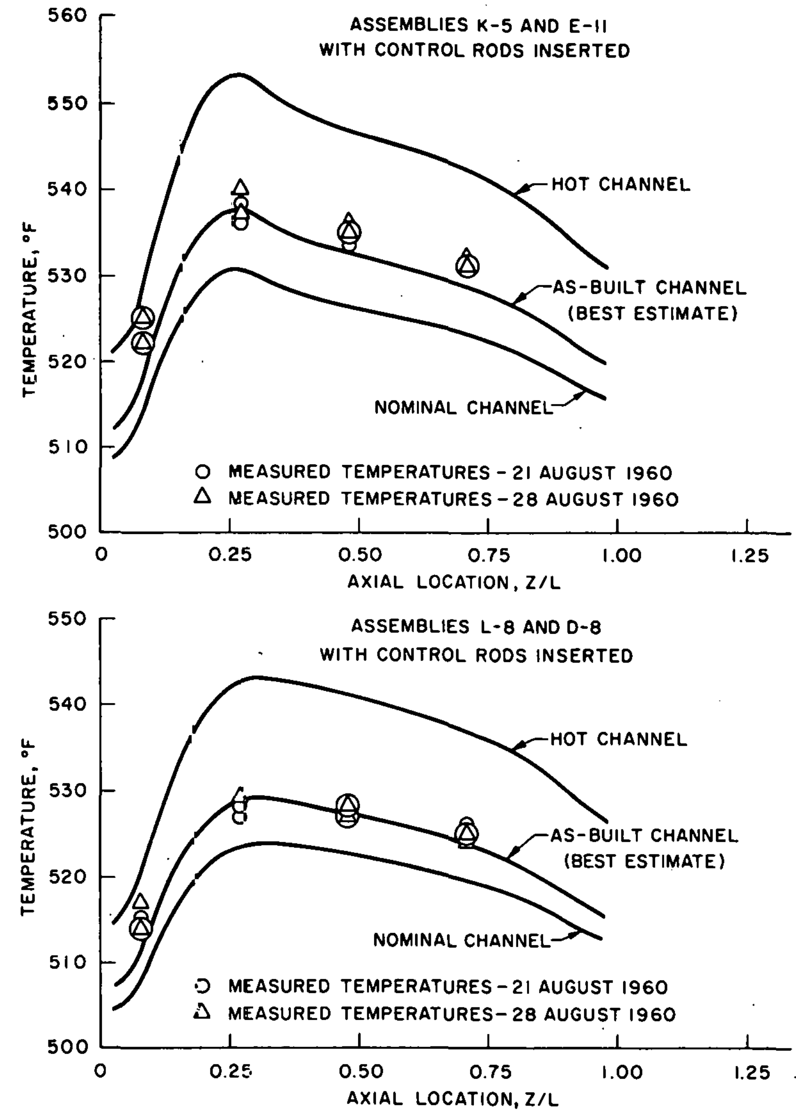


Figure III-9. PWR Seed Fuel Plate Temperature Predictions and Readings.

the upper and lower limit curves. In all cases the shape of the axial power distribution as indicated by the measurements agrees well with the predicted shapes.

The difference between nominal channel predictions and actual measurements may in part be attributed to: (1) the actual channel dimensions being slightly less, as found to be the case in an as-built core survey, than the nominal values used in the predictions; (2) the uncertainty in the measured reactor power level; (3) the uncertainty in thermocouple correction factors; and (4) the actual nuclear peaking factors being different from the values used in the predictions. The nuclear design data used represented pre-operational nuclear design predictions, indicating that the Group II control rods would be withdrawn to 21 inches at 50 equivalent full power hours (EFPH). Actually, the Group II control rod height of 21 inches was reached at approximately 1800 EFPH. The effect of his difference in reactivity change was not factored into the comparison because it is estimated to be negligible.

The agreement between the measured fuel plate temperatures and the predicted values, both in magnitude and axial distribution, is considered to be good.

Subsequent to the comparison of measured and predicted fuel plate temperatures described above, an analysis was made in which measured seed coolant channel dimensions applicable to the instrumented plates were used (rather than nominal values) in conjunction with a more sophisticated choice of nuclear factors. It was thus possible to increase the lower limit curve to a best estimate level representative of the actual data as shown in Figures III-8 and III-9. This modification was then considered applicable to all future predictions of instrumented plate thermocouple readings.

CHAPTER 4

CONTROL ROD DRIVE MECHANISM PERFORMANCE

Introduction

The control rod drive mechanisms position the control rods in the reactor. Each of the 32 mechanisms is completely independent mechanically; however, they are normally operated electrically, in synchronized groups of four, to maintain symmetrical flux distribution. These electrically powered mechanisms employ the principle of the jackscrew to move the control rods. The mechanisms completely release the control rod shafting, permitting a scram whenever power to the mechanism is interrupted.

The only mechanism component failure during Seed 2 operation was a defective stator in position E-12 in June, 1960. This stator was replaced with a spare, and the mechanisms operated during the remainder of Seed 2 life without incident. The operating experience, together with the periodic test data, indicates that the mechanisms performed their functions adequately and satisfactorily for the Seed 2 period.

Seed 2 Operating Experience

Stator Failure

The mechanism stator in position E-12 showed low insulation resistance to ground during a routine check on May 2, 1960, and again on May 4 and June 13. In each of the three cases, the resistance was increased sufficiently to permit mechanism operation by energizing the stator for a period of time to dry the winding. This indicated that the low insulation resistance was caused by the presence of moisture in the stator and stator cavity. The recurrence of the low resistance on June 13 indicated that the intervening period of operation had not removed the moisture and it was concluded that the stator cavity contained a quantity of water or that water was entering the cavity by leakage. On June 18, 1960, during a plant shutdown period, the stator winding showed a solid ground (less than 0.5 ohms) and further efforts to dry it out were considered futile. During this entire period, the mechanism had performed its function normally; however, it was decided to replace the stator assembly with a spare to prevent possible failure during operation.

The defective stator was replaced with a spare on June 21, 1960. During this installation, the motor tube was inspected carefully (visually) for leaks. No leaks were detected and the spare stator assembly was installed, electrical operation was checked, and the mechanism was released for operation. Insulation resistance checks made as a part of start-up procedures and periodic testing show that the mechanism operated normally following this stator replacement.

The defective stator was returned to BAPL and both helium and hydrostatic leak tests were performed on the water jacket. No leaks were detected but the stator showed a winding-to-core ground. The stator was then sectioned and the defect was found to be a break in the slot insulation. There was no evidence that faulty construction, defective materials, damage, or a condition other than the presence of moisture caused this failure.

The stator failure is believed to have resulted from an incident during refueling which caused a spray of cooling water in the vicinity of this mechanism while the watertight caps were off the electrical receptacles. The prolonged exposure to moisture apparently led to the deterioration and failure of the insulation.

The incident causing the wet stator during refueling could not occur during normal operation when plugs are installed in the receptacles; therefore, no design changes were made on the Core I mechanism. Additional care will be taken during future refueling operations to prevent a recurrence of the failure.

The Core II mechanism design will use sealed stator and sealed position indicator coil components, which will prevent insulation damage by water during any phase of refueling or condition of operation.

Precritical and Initial Tests (DLCS 1580202 and DLCS 1580302)

Precritical and initial tests were performed after the installation of Seed 2 to assure that the mechanisms and their associated instrumentation were functioning satisfactorily after refueling and prior to reactor start-up.

The initial operation test (DLCS 1580202) was completed on April 11, 1960. During this test the rod bottom indicator for mechanism position 81 was found to be inoperable because of interchanged connections. The connections were corrected and the indicator was retested with satisfactory results. Five mechanisms were found to have reversed rotation. These were reconnected for correct rotation and retested. After correction of these wiring errors, the position indication system, the power supply, and the control rod drive mechanisms were found to be satisfactory for operation.

The precritical operational tests (DLCS 1580302), consisting of insulation resistance test, latching and stationary scram test, rotational test, and scram time test, were completed on April 29, 1960. The mechanisms operated satisfactorily.

Periodic Tests (DLCS 1480113 through DLCS 1480116)

The mechanism periodic tests were performed four times during Seed 2 operation. These tests consist of insulation resistance test, latching and stationary scram test, rotational test, and scram time test, and are designed to furnish information permitting the evaluation of mechanism performance and condition. The test also reports the rod travel in feet and the number of scram operations, which provide an indication of the amount of service each mechanism has experienced. The data confirmed that the mechanisms were functioning satisfactorily and that there were no tendencies toward a significant increase in mechanism and control rod friction or deterioration of mechanism performance.

The total rod motion was reported as a part of this data and furnishes a measure of the mechanism and rod wear. The total travel is approximate, but is believed to be accurate to plus or minus ten percent. The total rod travel is shown in Table III-G for the Seed 2 operating period, as well as the total accumulated travel since the original start-up of the reactor.

The design requirements and engineering proof test acceptance limits were based on 10,730 feet of rod travel. Table III-G shows that the design requirements were conservative and that a third seed of operation is well within the design life capability of these mechanisms.

The total number of scram operations are also reported (Table III-H) as a part of this periodic test data. Scrams listed in Table III-H are control rod insertions from a height greater than 15 inches. The data presented in Table III-H show the number of scram operations and the number of times de-energized for Seed 2. The Seed 1 data and the accumulated totals are also shown for comparison purposes. The total scram operations experienced by these mechanisms to the end of Seed 2 life are approximately one-fourth the functional requirement, which is 201 scrams during the operating life of the mechanism.

Radiation Survey of the Reactor Vessel Head (DLCS 3050104 through DLCS 3050107)

The periodic survey is made of radiation levels on the mechanisms and other equipment on the vessel head. This is done to detect buildup of radioactivity in these components by accumulation of crud. The effect of rod exercise on radiation levels measured at the mechanisms was investigated during Seed 2 operation. Although a specific relationship between rod (mechanism) exercise and radiation level could not be obtained from these tests, the data indicate that operating a mechanism tends to reduce the radiation levels along the mechanism.

TABLE III-G

TOTAL ROD TRAVEL DURING SEED 1 - SEED 2 OPERATION

Subgroup	Seed 2 Total Feet	Seed 1 Total Feet	Accumulated Total Feet
10's	2559	1580	4139
20's	2289	1466	3755
30's	2547	1600	4147
40's	1459	1072	2531*
50's	2541	1614	4155*
60's	2155	2092	4247
70's	2542	1560	4102†
80's	1089	667	1756

* Mechanism serial nos. 822 and 827 were interchanged during refueling. Mechanism no. 822 operated in 40's subgroup (position L-8) during Seed 1 and in 50's subgroup (position L-7) during Seed 2. Mechanism no. 827 operated in 50's subgroup (position L-7) during Seed 1 and 40's subgroup (position L-8) during Seed 2. The totals are no. 822 - 3613 ft., and no. 827 - 3073 ft.

† Mechanism serial no. 837 in this subgroup (position E-4 was replaced by spare mechanism no. 814 during Seed 1 - Seed 2 refueling so each has accumulated only one seed period of use as shown in this tabulation.

TABLE III-H

TOTAL NUMBER OF SCRAMS DURING SEED 1 - SEED 2 OPERATION

Subgroup	Seed 2 No. of Scram Operations	Seed 2 No. of Times De-energized	Seed 1 No. of Scram Operations	Seed 1 No. of Times De-energized	Accumulated Total No. of Scram Operations	Accumulated Total No. of Times De-energized
10's	18	69	29	139	47	208
20's	11	70	30	115	41	185
30's	19	69	29	139	48	208
40's	7	70	10	111	17	181*
50's	19	69	31	139	50	208*
60's	11	69	27	113	38	180
70's	19	69	30	138	49	207†
80's	6	69	9	111	15	180

* Mechanism serial nos. 822 and 827 were interchanged during Seed 1 - Seed 2 refueling. Mechanism no. 822 operated in 40's subgroup (position L-8) during Seed 1 and in the 50's subgroup (position L-7) during Seed 2. Mechanism no. 827 operated in 50's subgroup (position L-7) during Seed 1 and in 40's subgroup (position L-8) during Seed 2. Accumulated totals are: no. 822 29 scrams, 180 total number of times de-energized; no. 827 40 scrams, 209 total number of times de-energized.

† Mechanism serial no. 837 in 70's subgroup (position E-4) was replaced with spare mechanism no. 814 during Seed 1 - Seed 2 refueling so that each has accumulated only one seed period of use as shown in tabulated columns.

Conclusions

The control rod drive mechanisms have functioned satisfactorily and reliably during Seed 2 operation. Periodic test data show that the accumulated rod travel during Seed 1 and Seed 2 operation is equal to approximately 40 percent of the design life value. The exercising of mechanisms appears to be a practical method of reducing the quantity of loose crud in the mechanism and, thereby, effects a reduction in radioactivity level in the reactor vessel head area.

PART IV

REACTOR PLANT PERFORMANCE AND MODIFICATIONS

CHAPTER 1

OPERATION AT STEADY-STATE LOADS

Introduction

Approximately 7350 EFPH of the 7900 EFPH lifetime of Core I Seed 2 were obtained during full power steady-state operation with an average primary coolant temperature of 500°F and a coolant pressure of 1800 psia. In addition, there was limited testing at load levels other than full power and at average coolant temperatures other than 500°F. Some of the significant results of these steady state operations are summarized below.

Operation at Full Power and 500°F T_{avg}

One hundred percent full power reactor thermal output is nominally defined as 231 megawatts. That value has been established as a power level sufficient to produce a net generated electrical output of 60 megawatts with three primary coolant loops in service during normal conditions of plant operation. During operation with four coolant loops in service, the required reactor thermal output is approximately four percent higher to account for the power requirements of the fourth main coolant pump. The actual reactor thermal output required to generate 60 megawatts net will, of course, vary as the parameters that affect station efficiency change. Similarly, when equipment is in other than the normal configuration, such as when one or more feedwater heaters have been removed from service, the required reactor thermal output will usually be greater.

Figure IV-1 presents a comparison of the reactor thermal outputs required during Seed 2 full power operation. Since the gross generated output at full power varied somewhat with the demands of the system and the station, the values of reactor thermal output have been normalized by dividing reactor thermal output by gross generated output. The resulting values of megawatts thermal per megawatt gross electrical are plotted in Figure IV-1 for various times during Seed 2 depletion. The primary coolant loops in service and the feedwater heaters out of service are designated on the illustration. As is evident, the reactor thermal output per generated megawatt was approximately 7 to 10 percent higher with one or more of the feedwater heaters removed from service. The variation of values for a given condition is attributed in part to the effect of river water temperature on turbine condenser vacuum, which, in turn, affects the efficiency of the turbine. In the case of the readings taken during April of 1961, the apparently high values are due to a combination of overconservatism in the method of calorimetric determination, and to the removal of two of the feedwater heaters from service.

The reactor thermal output is normally determined from measurements taken of feedwater flowrate and temperature and steam temperature. The enthalpy gain from feedwater to steam is multiplied by the feedwater flowrate to each heat exchanger to determine the power output of each coolant loop (a secondary calorimetric). During the full power run beginning April 14, the feedwater flowrate instrumentation on the lines to the individual heat exchangers required recalibration, thereby requiring the use of the readings of total feedwater flowrate, plus a three percent adjustment, in order to account, conservatively, for the previously observed differences between the two methods of

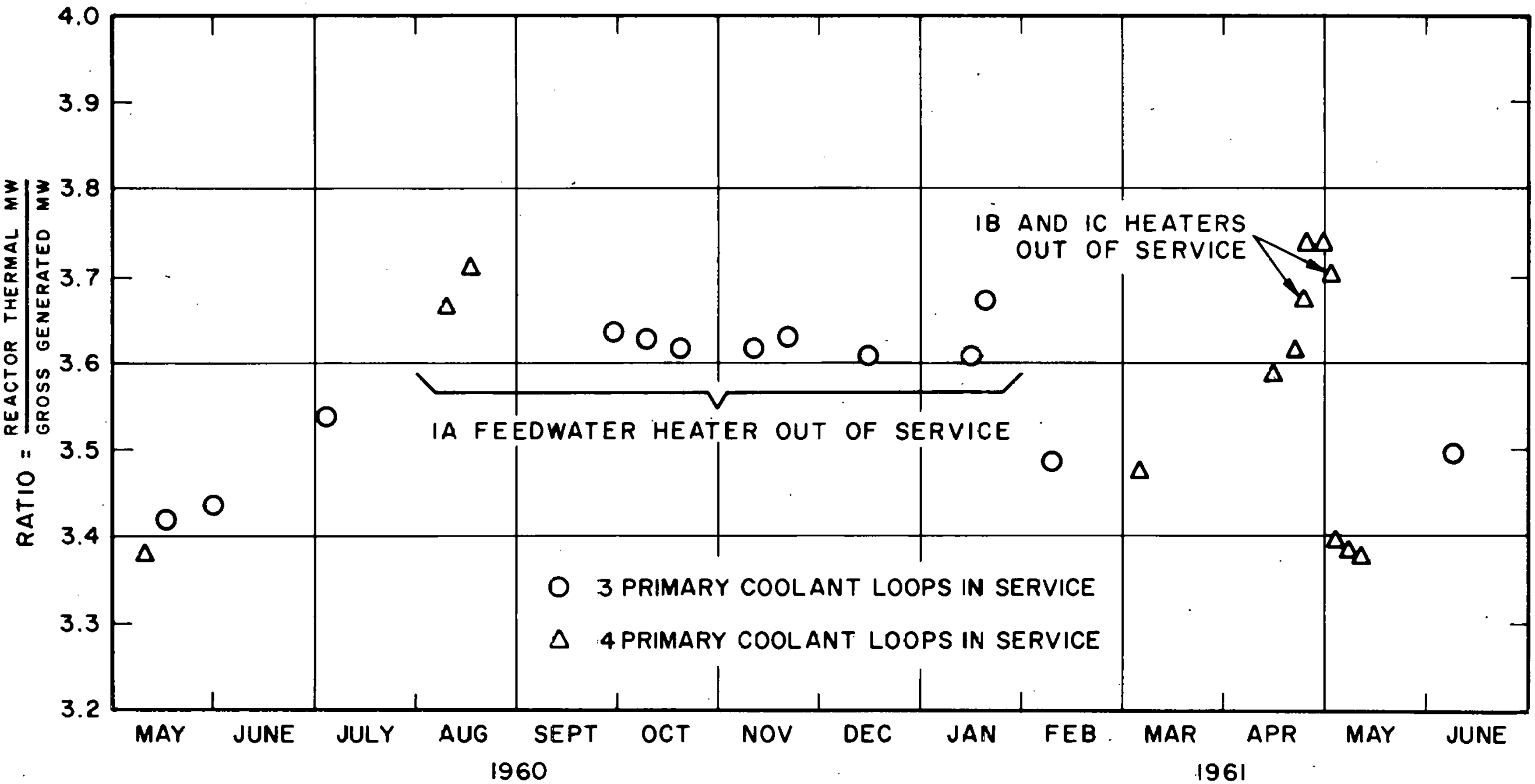


Figure IV-1. Megawatts of Reactor Thermal Output Required per Megawatt of Gross Generated Output at Various Times During Core I Seed 2 Full Power Operation.

determination. This method was used until a jump in the total feedwater flowrate readings, later attributed to leakage in the 1B feedwater heater, resulted in uncertainty in this second method of determination. Beginning April 17, the reactor thermal output was determined from readings of primary coolant temperature entering and leaving the reactor and from primary coolant loop flowrate (a primary calorimetric). The resulting values were conservatively increased by a nominal 8 percent to account for uncertainty in the method and to account for previously observed differences between primary and secondary calorimetrics. Since the 1B and 1C feedwater heaters were removed from service during this same period in order to repair the leaking B heater, the combination of reduced efficiency and conservation in calculation resulted in the high values of reactor thermal output evident in Figure IV-1. The leakage in the 1B feedwater heater was corrected and the individual line feedwater flowrate instrumentation was calibrated and back in service by May 3.

Operations at Various Power Levels and Various T_{avg}

Limited testing was performed during Seed 2 depletion at steady-state power levels other than full power and at average coolant temperatures other than 500°F. Some of the results of this testing are presented in Figures IV-2 through IV-4.

Figure IV-2 shows the reactor thermal output required to generate various values of gross electrical output at an average coolant temperature of 500°F. The results plotted were obtained during two different performances of test DLCS 34901 at different conditions of turbine-generator efficiency. In the second test the 1A feedwater heater was out of service and high river water temperature resulted in reduced efficiency. In the first performance, the efficiency was more typical of normal operation. Although the two plots in Figure IV-2 form a band typical of most of Seed 2 operation, there were some periods of operation outside the band, both above and below.

Figure IV-3 shows plots of coolant temperature entering and leaving the reactor at various power levels during operation at average coolant temperatures of 480, 490, 500, 510, and 520°F. Since a primary calorimetric consists of the product of specific heat, coolant flowrate, and the difference between the coolant temperature entering and leaving the reactor, and since the first two parameters are nearly constant, the temperature difference between the coolant entering (Tc) and leaving (Th) the reactor may be utilized as a relative indication of reactor thermal output of the various test conditions. Examination of the differences (Th - Tc) at the various T_{avg} for a given gross generated output indicates that a larger reactor thermal output was required at the lower values of coolant temperature. This is as predicted because of the reduced efficiency at the lower steam pressures resulting from lower T_{avg}. Some of the data plotted in Figure IV-3 were obtained with all of the feedwater heaters out of service, while the remaining points were obtained with all of the heaters in service. As would be expected, the reactor thermal output required was greater with the feedwater heaters out of service. The throttle steam pressures observed during the operations at various power levels and various coolant temperatures are plotted in Figure IV-4.

Heat exchanger thermal conductance, U, was investigated during the performances of test DLCS 34901 by calculation of the term UA (A = heat transfer surface area of heat exchanger). The average values of UA for the four heat exchangers varied from approximately 7.9×10^6 BTU/hr - °F at a 19 Mw power level to approximately 9.0×10^6 BTU/hr - °F at full power. The general upward trend with increasing power level was as would be expected. The UA's at full power were within the band of predicted full power UA nominal values of from 9.6×10^6 BTU/hr - °F for clean conditions to 7.5×10^6 BTU/hr - °F for fouled conditions (0.0003 hr-ft^2 - °F / BTU fouling resistance).

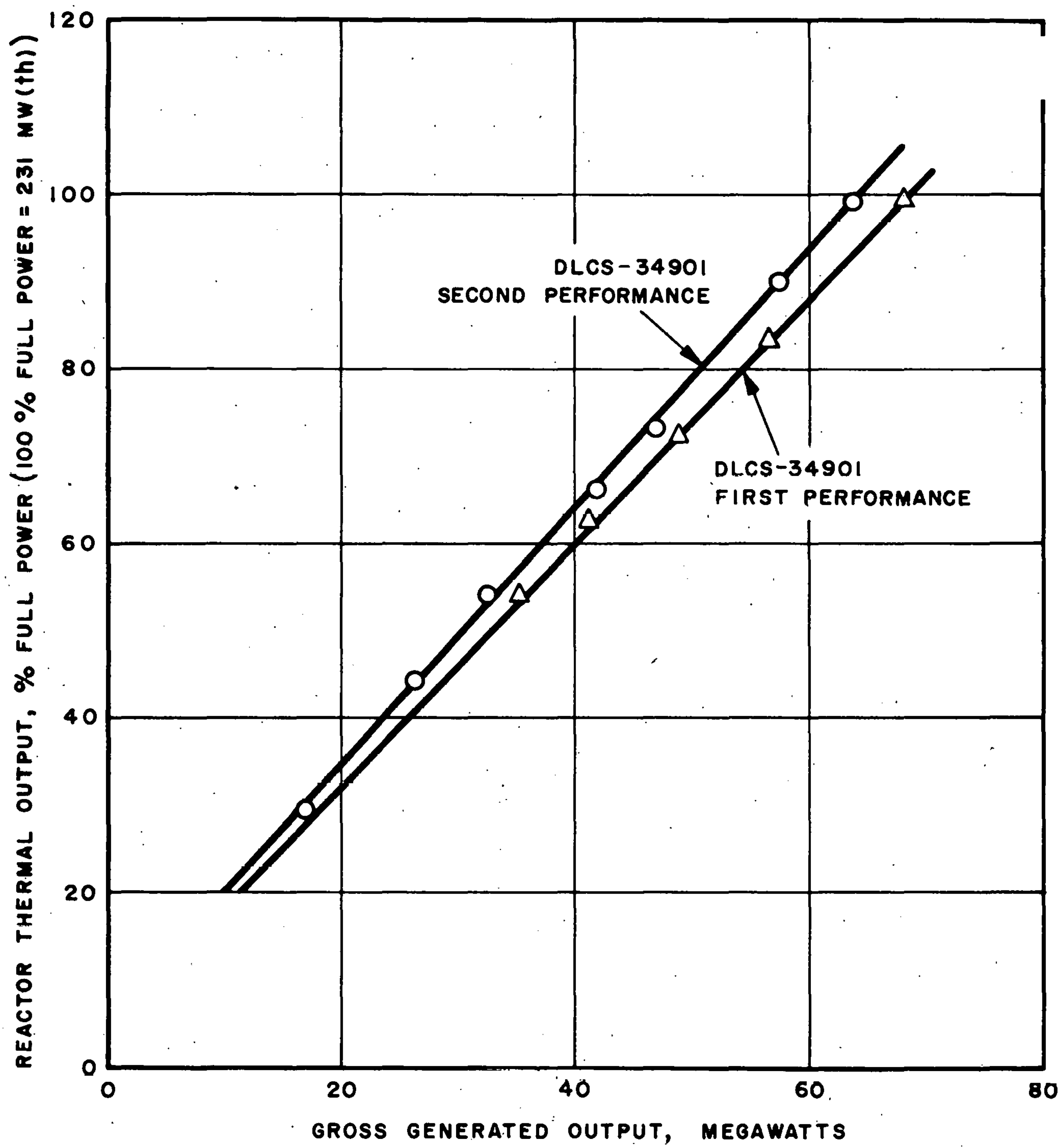


Figure IV-2. Reactor Thermal Output vs Gross Generated Output During First and Second Performances of DLCS 34901.

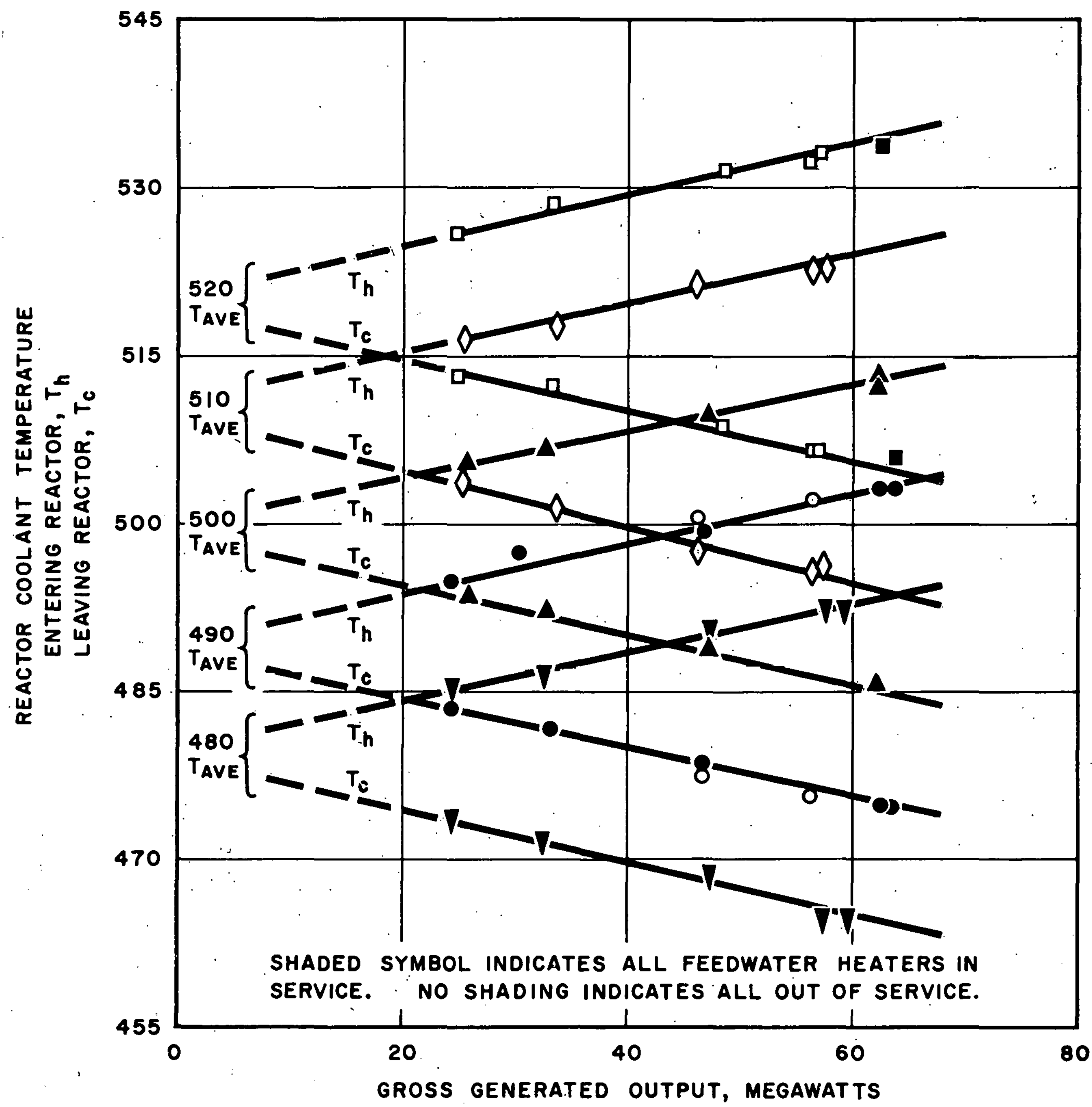


Figure IV-3. Reactor Coolant Temperature Entering and Leaving Reactor at Various Power Levels and at Various Conditions of Average Coolant Temperature During Test DLCS 3690101.

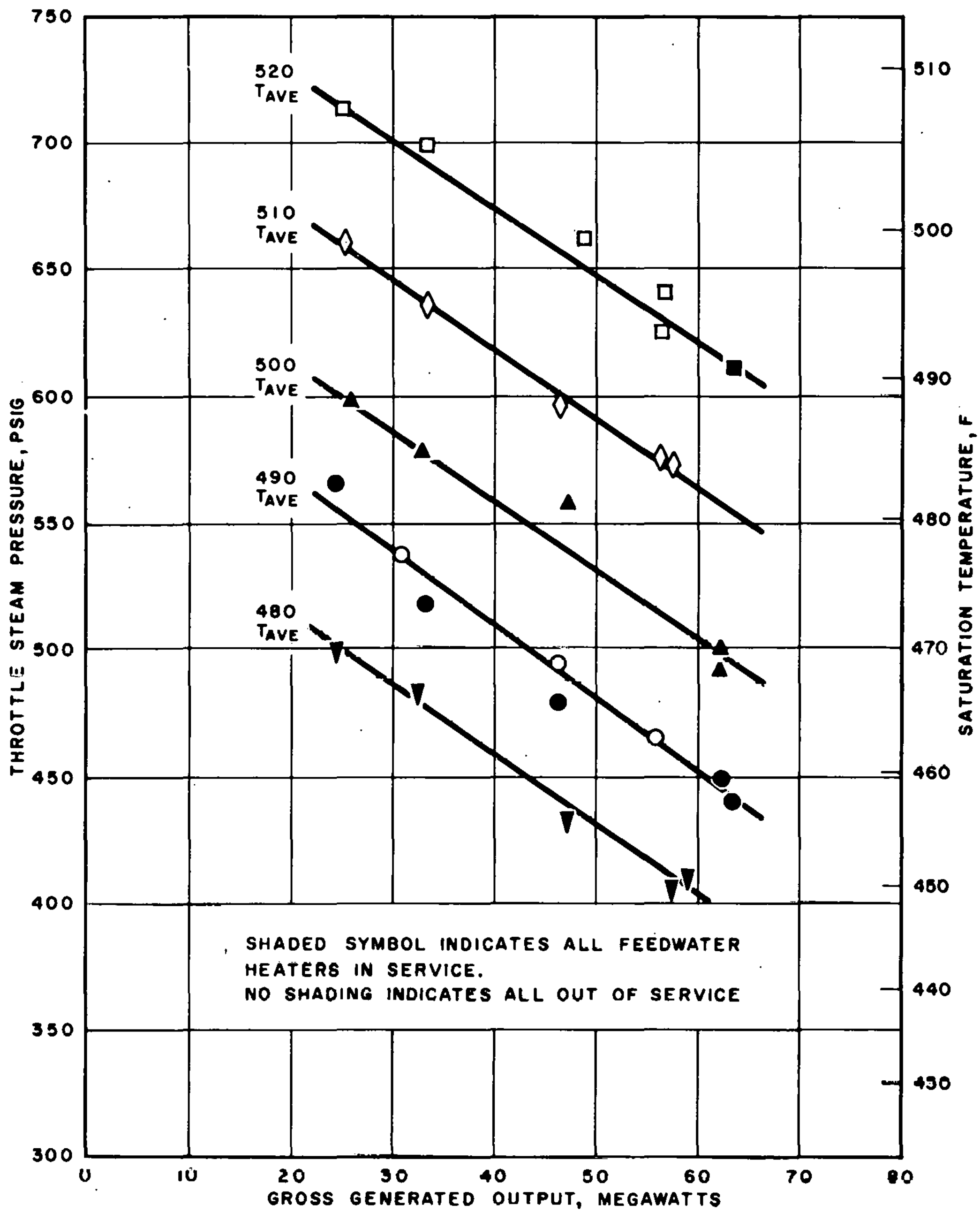


Figure IV-4. Throttle Steam Pressure and Saturation Temperature at Various Power Levels and at Various Conditions of Average Coolant Temperature During Test DLCS 3690101.

Conclusion

In summary, steady-state operation during Seed 2 depletion demonstrated the ability of the reactor plant to produce sufficient power to generate 60 megawatts of net electrical output, even during conditions of high river water temperature, with all of the feedwater heaters in service. Because one or more feedwater heaters were removed from service during much of Seed 2 operation, the average net generated output was slightly less than 60 megawatts. During conditions of operation at load levels other than full power and at average coolant temperatures other than 500°F, the reactor plant and steam plant parameters varied as would be expected.

CHAPTER 2

OPERATION DURING TRANSIENT CONDITIONS

Introduction

The two types of transient conditions that the Shippingport reactor plant may be expected to undergo are generator load transients and control rod reactivity transients. These transient conditions cause excursions in reactor power level, coolant temperature, and coolant pressure. Tests were performed during Seed 2 operation to confirm the adequacy of the plant during generator load and control rod reactivity transients. The test results indicate that the plant responds smoothly and is similar for both Seed 1 and Seed 2.

Generator Load Swing Transients

Generator load transients cause temperature and pressure excursions in the reactor coolant. During load swing transients the reactor plant is controlled inherently by the negative temperature coefficient of reactivity. Manual insertion or withdrawal of control rods is employed to supplement the temperature coefficient. Reactor coolant pressure is controlled by the pressurizer, pressurizer pressure regulating heaters, and the pressurizer spray.

During normal operation, the generator load does not change rapidly. The most severe load transients the Shippingport Atomic Power Station was expected to accommodate are:

1. Plus 15 or minus 12 Mw at a step change rate.
2. Plus or minus 15 Mw at a rate of 3 Mw per second.
3. Plus or minus 20 Mw at a rate of 25 Mw per minute.

These load swings are applicable within the full power range and are based on the system capacity and loading rates. Generator load swing tests were performed to provide data on the reactor plant transient response. The adequacy of plant response is illustrated in Figures IV-5 and IV-6 for a load decrease and a load increase, respectively. These figures illustrate, as a function of time, the variation of load, nuclear level, average coolant temperature, and coolant pressure. In each figure a comparison is made between a ramp change in load and a step change in load. (The step change in load was effected by opening steam relief valves.) The transients in Figures IV-5 and IV-6 are for load changes of approximately 20 to 25 Mw. The average coolant temperature and pressure excursions were controlled by the negative temperature coefficient and pressurizer; no external control (rod motion or pressurizer spray) was employed.

Though the step and ramp load changes are not directly comparable, they do serve to illustrate that the step load change does not necessarily cause transient excursions more severe than the fast ramp change. This differs from Seed 1 data which indicated that the more rapid rate of load change caused greater excursions. However, it was noted in the Seed 1 operations report that the reverse could be true at other rates and magnitudes of load change. Inspection of Figures IV-5 and IV-6 reveals that the front edge of the average coolant temperature excursion is steeper for the step load

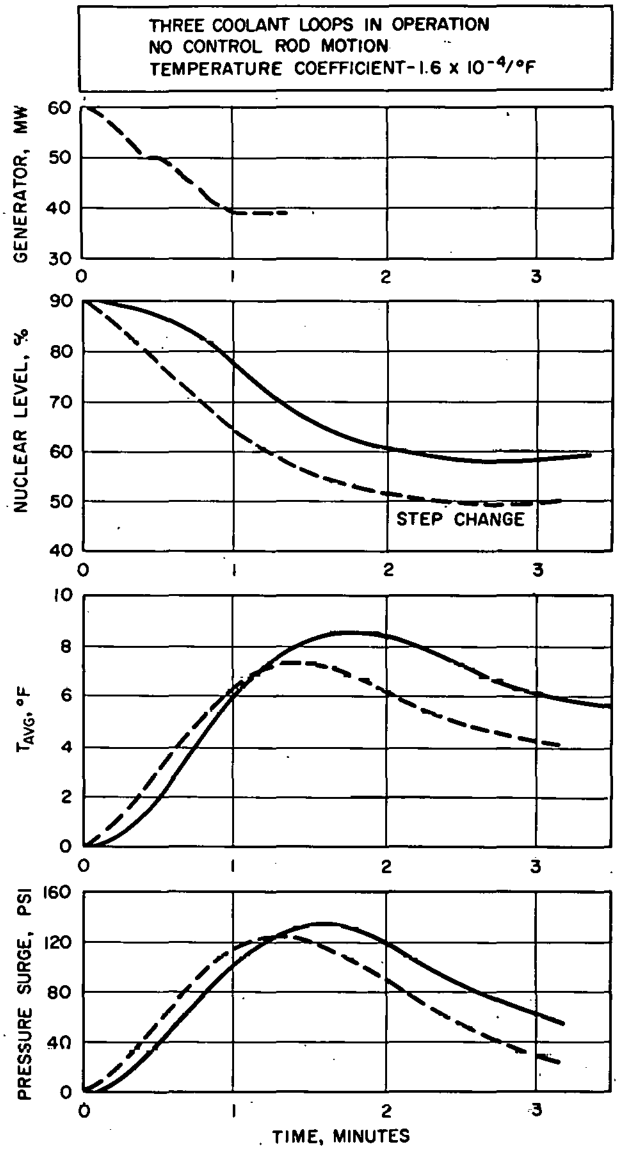


Figure IV-5. 21 MW Load Reduction - Comparison Step and Ramp Change.

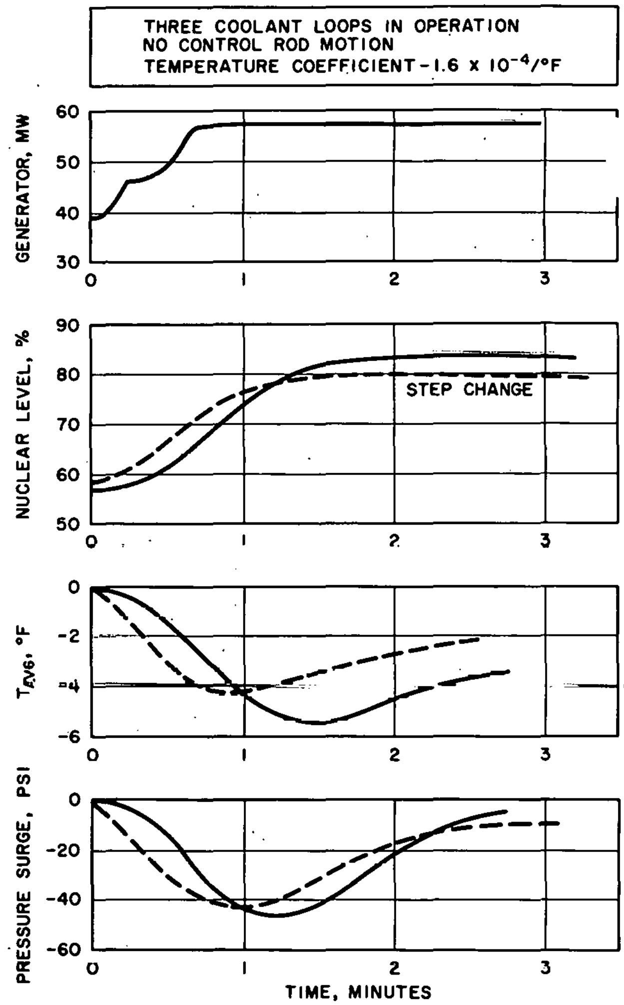


Figure IV-6. 18 MW Load Increase - Comparison Step and Ramp Change.

change than for the ramp load change. Hence, temperature coefficient reactivity in initially larger for the step change and the nuclear level change is more rapid. This results in a faster transient and smaller excursions for the step change of load than for the ramp change of load.

The effect of automatic operation of the pressurizer spray on the pressure and temperature excursion for a step load reduction is illustrated in Figure IV-7. The pressurizer spray valve is pressure actuated and opens at about 1870 psia (70 psi above operating pressure) and closes at about 1800 psia. Figure IV-7 illustrates a reduction in the peak pressure surge from 170 psi with no spray to 90 psi with automatic spray for a 35 Mw step load reduction. However, with pressurizer spray operation there is a subsequent negative pressure surge. This result is due to the pressurizer spray condensing part of the steam bubble in the pressurizer. After the average coolant temperature peaks and then decreases to steady state, the volume available for the steam bubble increases and allows the steam bubble to expand. Since the mass of the steam bubble has been reduced, a negative pressure surge results.

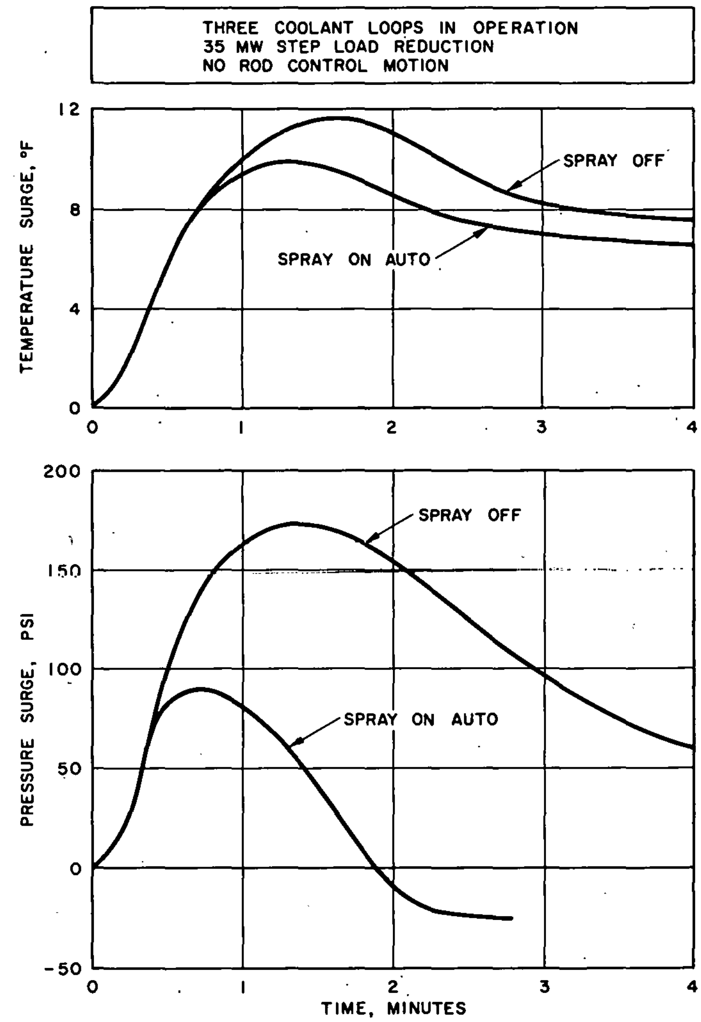


Figure IV-7. Effect of Pressurizer Spray on Plant Transients.

A second salient feature illustrated in Figure IV-7 is the reduction in the peak average coolant temperature excursion with operation of the pressurizer spray. This is due to reduction in the pressure coefficient reactivity. The pressure coefficient of reactivity is positive and, hence, subtracts from temperature coefficient reactivity. Reducing the pressure increases the effectiveness of temperature coefficient reactivity and the nuclear level adjusts more rapidly to the load demand. This reduces the integrated net difference between the load and nuclear level, and consequent smaller temperature excursions occur.

During the depletion of Seed 2, Shippingport was operated at full power for extended periods up to 1300 hours. Each of these extended power runs was terminated with a rapid station shutdown which provided operational data on generator load transients. The transient consisted of reducing the generator load from about 68 Mw to zero Mw as rapidly as possible, utilizing the main governor control (approximately 1.5 minutes). The generator output, coolant temperature excursions, and coolant pressure excursions for rapid station shutdowns, performed at two times in Seed 2 life, are presented in Figures IV-8 and IV-9. Both manual control rod motion and manual operation of pressurizer spray were utilized in controlling the reactor coolant temperature and pressure transients. There transients serve to illustrate the maneuverability of the reactor plant, with the application of external control features.

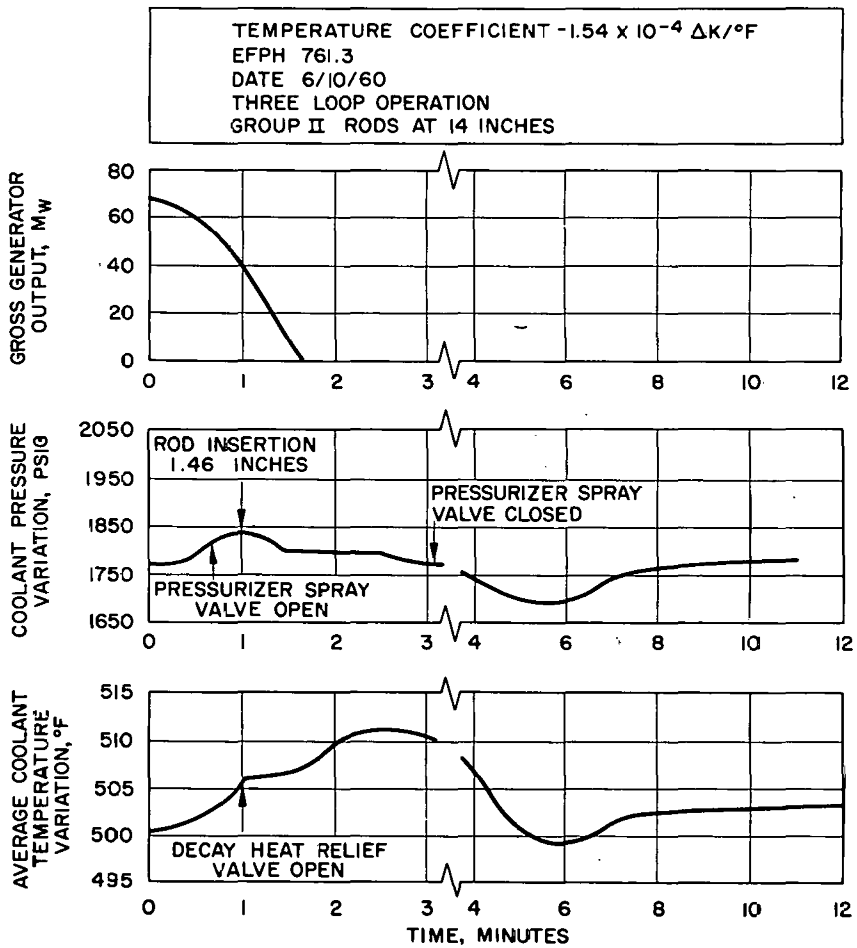


Figure IV-8. Variation of Plant Parameters during a Rapid Station Shutdown from Full Power Operation (EFPH 761.3).

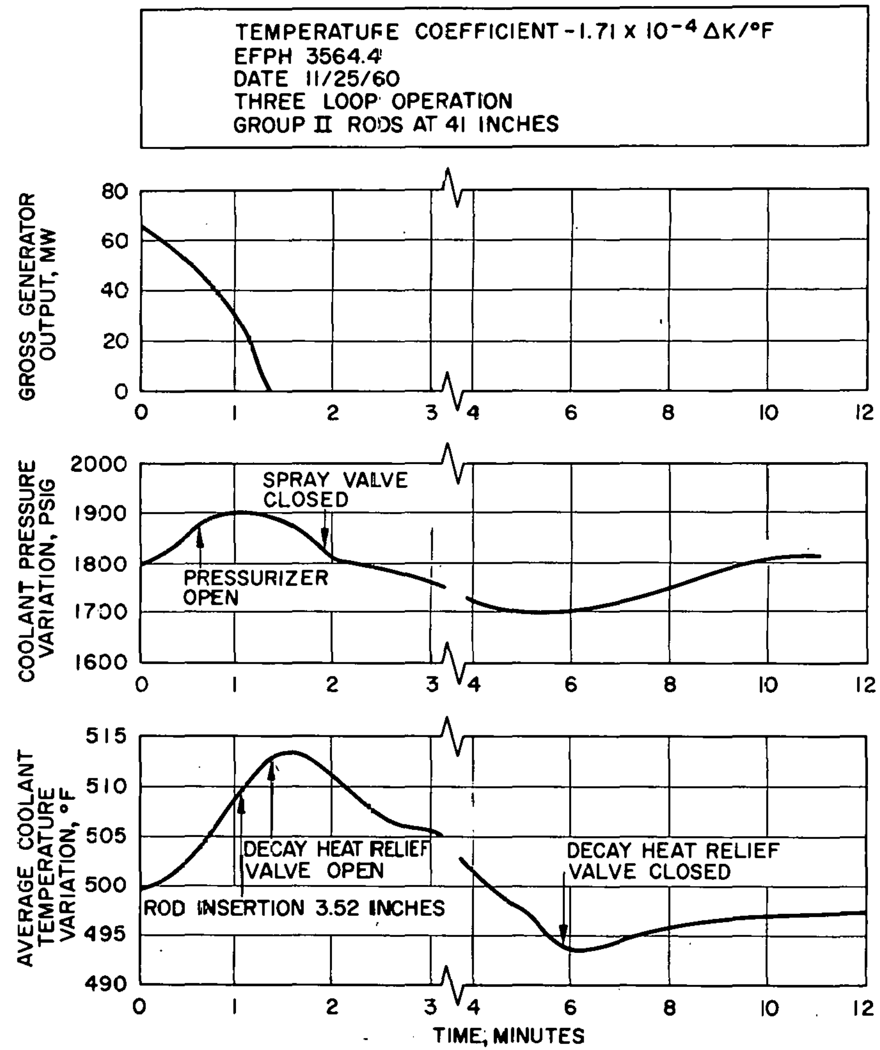


Figure IV-9. Variation of Plant Parameters during a Rapid Station Shutdown from Full Power Operation (3564.4 EFPH).

Control Rod Reactivity Transients

Rod reactivity transients result when control rods are continuously withdrawn and the reactor is made supercritical. Rod reactivity transients may be categorized into two groups, according to the power level of the reactor at the initiation of rod withdrawal. Start-up transients occur when the reactor is made supercritical while in the subpower range and the power level is allowed to increase and enter the power range. Power range rod withdrawal transients occur when rods are withdrawn continuously with the reactor plant operating at some steady-state power level. In both types of rod withdrawal transients, the result is a nuclear power excursion with attendant average coolant temperature and pressure excursion. Such transients are not allowed to continue or the thermal margin of the reactor core may be exceeded. Automatic shutdown of the reactor core on high power level or high reactor coolant temperature protect the core and limit the extent of rod withdrawal transients.

Typical response of the reactor plant to a start-up transient is illustrated in Figure IV-10 which presents the variation with time of nuclear level, hot leg temperature, average coolant temperature,

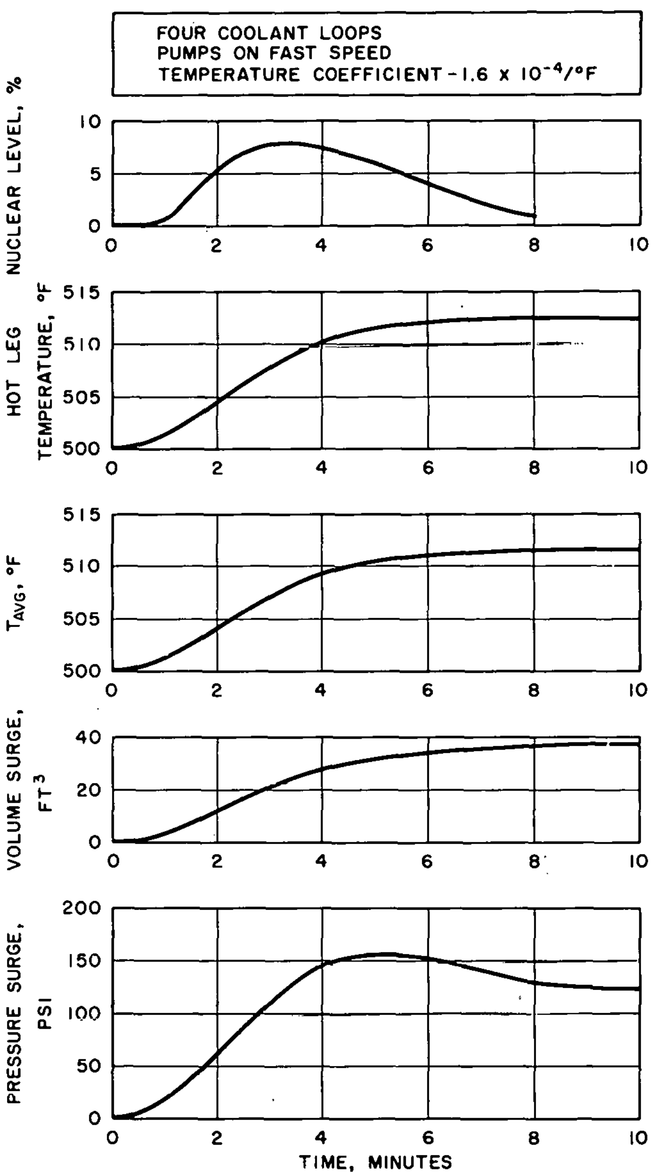


Figure IV-10. 0.94 DPM Start-up Transient.

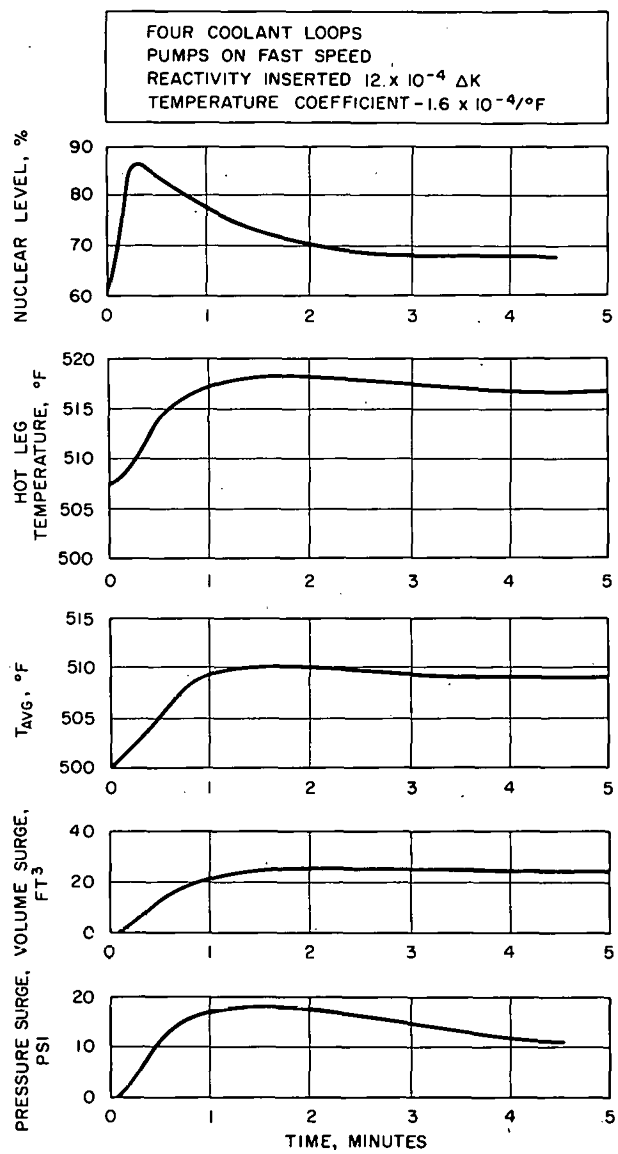


Figure IV-11. Rod Withdrawal Transient from 60 Percent Power.

coolant volume, and coolant pressure for a start-up transient of 0.94 decades per minute. It is noted that after having withdrawn control rods to initiate the transient, no external control was employed. The transient is the result of the initial rod motion and the temperature coefficient acting to restore steady-state. The transient is a mild transient, the nuclear level peaking at about 7 percent, and is typical of transients occuring when the reactor is utilized to warm up the reactor plant. The transient results in about a 10°F rise in average coolant temperature. This rise in temperature increases the coolant volume about 30 cubic feet and causes a pressure surge of about 160 psi. Of course, if the reactor was being used to warm up the reactor plant, approximately 30 cubic feet of reactor coolant would be discharged to the flash tank and the pressure surge would be greatly reduced.

The response of the reactor plant to a rod withdrawal transient while operating at a steady-state power level of 60 percent is illustrated in Figure IV-11. In this transient the control rods were continuously withdrawn about 0.75 inch, inserting about 1.2×10^{-4} reactivity. The figure illustrates that the nuclear level rises rapidly from the steady-state power level of 60 percent to a peak power level of 85 percent in about 20 seconds. The excess heat increases the reactor coolant temperature and produces a peak rise of 10°F in average coolant temperature. The increased temperature expands the reactor coolant and a consequent pressure surge of 180 psi results.

In actual practice, control rods are not withdrawn continuously when the reactor is critical, but rather in small increments. This transient, performed as part of testing on Seed 2, serves to illustrate the results of a continuous rod withdrawal transient.

<u>Summary</u>

Operations and test data from Shippingport during Seed 2 depletion have indicated transient response similar to that observe during Seed 1 operation. The effectiveness of the negative temperature coefficient in adjusting the reactor power level to load demands was adequate as was illustrated. The maneuverability of the reactor plant was demonstrated by the rapid station shutdown transients.

External control of the reactor plant has proved very effective. The pressurizer spray has the capacity to reduce by one-half the pressure excursion resulting from large load reductions. The effect of control rod motion varies, depending on the rod worth and the time in the transient at which it is employed. The external controls available to the reactor operator are adequate.

CHAPTER 3

SYSTEMS AND STRUCTURAL EXPERIENCE AND MODIFICATIONS

Introduction

The reactor plant during operation of Core I Seed 2 operated at a nominal coolant pressure of 1800 psia and an average coolant temperature of 500°F. The plant operated on four main coolant loops (when available) although only three loops are required to produce the design generated output of 60 megawatt net electrical. When operating on only three loops, the fourth loop was out of service for the following reasons:

1. Repair and leak testing of the 1A Steam Generator.
2. Main coolant pump testing and gasket repair.

The reactor plant hydraulic systems operated satisfactorily with the exception of the limitations on the operation of the Reactor Plant Cooldown and Temperature Control System and a short period during the summer of 1960 when the cooling capacity of the reactor plant component cooling system was marginal. The former system could not be placed in high-temperature service because of a plant limitation restricting the temperature differential in the steam generator blowdown lines.

During operation of Seed 2, no electrical problems developed which resulted in prolonged station shutdown or which caused serious operating difficulties. The improved performance of electrical systems and components over that experienced during Seed 1 operation is primarily attributed to the fact that the major problem areas were identified and corrected prior to initial operation of the second seed. There were, however, a number of minor problems, both chronic and unique. In addition, a number of modifications were made to the various systems in order to improve their performance.

Modifications, alterations, and additions to the plant have continued throughout the Seed 2 life operating period.

Mechanical and Structural Changes

Several changes contributing to the improvement of the reactor plant operation and maintenance were made during this period of plant operation. These changes were:

Pressurizer Vessel Surge Line Temperature Differential Limitation

The allowable temperature differential across the pressurizer surge line was increased from 200°F to 400°F during a plant warmup only. This change permits drawing a steam bubble in the pressurizer vessel with the reactor coolant temperature as low as 50°F and the primary pressure high enough (300 psig) to permit operation of the main coolant pumps. Reactor heat may now be utilized for plant warmup over a wider temperature range (50°F to 500°F, where formerly it had been 223°F to 500°F) which permits a shorter plant warmup time.

Chemical Addition Poison

The use of boric acid in the coolant chemical addition system for chemical shutdown has been replaced with potassium tetraborate (PTB). The PTB will not form a hard cake during storage; it i more soluble than boric acid; and it will not affect Malcomized 17-4 PH and stressed 44CC material (if injected as solution into the plant), whereas these metals are severely attacked by boric acid.

Reactor Plant Cooldown and Temperature Control System

Installation of the cooldown system was completed during this period. The system has been operated satisfactorily as follows:

1. Boiler water in laid-up cold steam generators is recirculated to provide representative boiler water samples.

2. The reactor plant can be cooled down from 250°F to ambient temperatures (solid water system).

3. The reactor plant temperature can be controlled below 250°F within ± 1°F (solid water system).

Reactor Plant Canal Water System

1. Demineralizer backflushing.

   As a result of experience with the canal water system, it was found feasible to backflush high activity insolubles from the dirty resin bed to the radioactive waste disposal system resin storage tanks (instead of replacing the resin with fresh resin) using temporary hose and an improvised procedure. The backflushing procedure was subsequently incorporated into the plant operating procedures. A permanent piping and valve change was made replacing the temporary arrangement.

2. Sample Station

   During refueling of Core 1 with Seed 2 the canal water system demineralizers became contaminated and required the use of temporary shielding around these vessels. New sample lines added to the system permitted sampling of the inlet and outlet water to the demineralizers at a sample station in a low radioactivity location.

Lithium Hydroxide Charging

A new lithium hydroxide line was added to the chemical addition system. This line permits injection of the lithium hydroxide solution into the coolant charging system without resulting in a coolant charging system conductivity alarm or contaminating the demineralized water supply to the chemistry laboratory.

Reactor Plant Component Cooling Water System

1. Temperature control.

   A small bypass line containing a throttle valve was added to the reactor plant component cooling water system on the river water side of the system's component cooler. The new bypass will permit fine adjustment of the component cooling water temperature by regulating the rate of river water through the cooler.

2. Flow measurement.

   Permanently installed flow measuring instruments were added to the reactor plant component cooling water system in normally accessible low radiation areas. These flowrates formerly were metered with a portable instrument and required personnel to enter restricted radiation areas.

3. Temperature alarm.

   A thermocouple for high temperature alarm (90°F) and indication of component cooling water heat exchanger outlet temperature was connected to the temperature monitor in the control room to permit better control of the component cooling water temperature.

4. Low-pressure alarm.

   A low-pressure alarm was added that signals, in the control room, a loss of system circulating pump discharge pressure.

Coolant Charging System

1. Fill pump recirculation.

   A small flow recirculation line was added to the coolant charging system fill pump to permit the fill pump to operate at high discharge heads without overheating and damaging the pump.

2. Temperature indicators.

   Local temperature indicators were added to the fill lines to each main coolant loop to aid the operators in maintaining temperature limitations during loop filling.

3. Charging pump relief valves.

   A minor change was made to the location of the coolant charging system charging pump relief valve discharge piping, which permits the relief valves to be isolated from the reactor plant water storage tank without isolating the tank from the reactor plant. This change will permit maintenance of either relief valve during plant operation.

Main Coolant Loop Drain Seals

The tell-tale drains located between the main coolant loop hydraulically and manually operated main stop valves were provided with loop seal piping. The loop seals are connected to the tell-tale drain valves with temporary piping. The loop seals prevent drainage of the main coolant loop pipe sections located between the hydraulic and main stop valves when a loop is isolated and the tell-tale valves are open. The loop seals prevent air from entering the main coolant loop, but allows any leakage from the high-pressure side of the hydraulically operated main stop valves to drain to the reactor plant container drainage system.

Reactor Plant Air Cooling System

The primary function of the reactor plant air cooling system is to provide forced ventilation to the container in order to maintain ambient air temperature of approximately 122°F. The system had an original design capability to remove 886,000 BTU/hr from the container with all four fans operating and an average reactor coolant temperature of 523°F. Subsequent experience indicated that the net amount of heat added to the container air was in the order of 1,473,500 BTU/hr with an average reactor coolant temperature of 523°F. Design data for Core 2 requires a heat removal rate of 1,688,000 BTU/hr with average reactor coolant temperature of 536°F, four loops in service and an average container air temperature of 122°F. Therefore, the system was redesigned for capability of 2,113,200 BTU/hr, which is approximately 25 percent over the expected net heat. During the Seed 2 - Seed 3 refueling period, actual modifications to the air cooling system will be completed and include the following changes:

1. Higher horsepower rated motors are to be installed; the existing fan sizes will be retained, but fan speeds will be increased.

2. Six air cooling units will be installed in the reactor plant container chambers to cool recirculated air.

3. Well water will be supplied to the cooling units.

The butterfly-type shut-off valves installed in the supply and exhaust portion of the air cooling system have not provided optimum leak tightness over extended periods, and, therefore, during the Seed 2-3 refueling period, the valves were returned to the manufacturer for rebuilding and modification of the gasketed seat design.

Reactor Plant Container

Access to the reactor plant container is through air lock type doors. The original door seals required excessive maintenance to maintain reasonable leak tightness. In an effort to maintain a more satisfactory seal at the edge of the airlock doors, the following changes have been made:

1. An air supply has been provided to pressurize the door gaskets automatically to force them into their seated position when the door is closed.

2. The shaft penetrations of all airlock operating mechanisms have been repacked with a chevron-type packing to reduce leakage at these joints.

3. Periodic door gasket lubrication has been incorporated into the maintenance schedule.

A circular window has been installed in the escape hatch of each reactor plant container door to permit visual observation into the airlocks. Emergency lighting has been provided within the airlocks for illumination in case of power failure. The existing electrical interlock system installed on the doors of airlocks no. 1 and 6, to prevent the operation of these doors if its companion door is not secured, has been modified to provide an audible alarm. This alarm signals that an improper airlock access operating procedure is being used.

Reactor Service Building Modifications

Experience during Seed 1 - Seed 2 refueling indicated the need for expanded facilities in the reactor service building. The three-shift per day operation carried out by over one hundred maintenance personnel illustrated the need for a larger locker room and associated clothing storage facilities. In order to provide for the increased activities associated with health physics, a field office, larger counting room, film calibration area, and administrative office were also required. An assembly room which could be used for instruction and as a lunch room was also desired. These additional facilities have been provided by internal modification of the original building and the construction of an extension on the southern end with a floor area of 1978 square feet.

Contaminated Equipment Room

Preparations for refueling of the reactor require an assembly inspection and packaging of all required tools and equipment prior to the start of the operation. All required items must be cleaned, identified, and placed in specific packages in order to carry out the sequential operations without delay. A temporary building placed on top of the 1BD boiler chamber concrete enclosure was used during Seed 1 - Seed 2 refueling. However, cleaning facilities were not available in this building. Therefore, this operation had to be performed in a converted warehouse located outside the plant boundary. This remote location was inconvenient and resulted in inefficiencies and delays. In addition, it is considered necessary to remove the temporary building to permit access to the boiler chamber during Core 1 - Core 2 modifications. In order to consolidate refueling preparation activities and provide additional laydown and assembly area, a building was erected adjoining the southwest corner of the fuel handling building. This building of 6000 square feet floor area contains a cleaning room, contaminated tool room, and laydown area; it is connected by a rolling steel door to the fuel handling building canal. Facilities are provided for direct receiving of truck deliveries of material for use in the reactor plant.

Construction Personnel Locker and Change Room Facilities

Preliminary estimates of manpower requirements for Core 2 plant modification activities indicate that as many as 80 men will be working within the reactor plant containers per shift on a three-shift per day basis. These major modifications within the containers are associated with the removal of the four existing boilers and reactor coolant pumps and installation of larger units. In parallel with this work, Core 1 will be removed and Core 2 installed. In order to maintain the necessary degree of control of personnel, tools, cleanliness, security, and to minimize interference, it is necessary to provide additional locker and change room facilities.

The existing reactor service building locker room will be occupied by the personnel associated with removal of Core 1 and installation of Core 2. No outside ground area of sufficient size to accommodate the proposed change room was available in a location convenient to the reactor plant container access openings. Therefore, the change room facilities were installed prior to Seed 2 - Seed 3 refueling on the floor of the concrete enclosure surrounding the auxiliary chamber. These facilities were used to complete certain portions of Core 2 modifications performed during Seed 2 - Seed 3 refueling. With this arrangement, it was possible for personnel to change clothing, smoke, and eat, all within this enclosure. The need to pass between this area and out-of-doors was necessary only at the start and end of each work day.

Nuclear Instrument Repair Shop

The size of the original instrument repair shop located in the basement of the turbine plant has proven inadequate in terms of size, proximity to work areas, and facilities. To overcome these deficiencies, an area was established in a former classroom located near the entrance to the turbine plant. This area is convenient to the "normal" work area, i.e., the control room, data acquisition equipment, the chemistry and physics counting room, and the entrance to the reactor chambers. This area was air conditioned to eliminate problems associated with maintenance of electronic equipment.

Mechanical and structural changes have also been made in anticipation of complete modification of the plant for future increased power generation capacity.

Heat Dissipation System

Core 2 will have a gross output equivalent to 150 Mw electrical. The nominal capability of the existing turbine-generator is 100 Mw. In order to dissipate the energy from the steam generated by the boilers, which is above this rating, it will be necessary to provide a condensing system. Therefore, a heat dissipation system is presently being installed which will receive steam from the four boiler drums at a "T" connection to the supply line to the existing turbine throttle valves. This steam will be condensed in two "U" tube type vertical condensers. The condensate is passed through a deaerator and returned to the boilers by means of two boiler feed pumps which will operate in parallel with the existing turbine boiler feed pumps. Cooling water for the condensers is to be supplied by two circulating water pumps, which take suction from a sump that is to be interconnected with the existing turbine condenser river water discharge tunnel. The circulating water from the heat dissipation system condensers will be returned to the river through discharge piping that ties into the existing discharge tunnel before the waste disposal effluent-circulating water mixing weir. In this manner, no new screenhouse intake structure is needed at the river. The heat dissipation system is of outdoor construction with only the steam driven circulating and feed pumps and control panels enclosed. The design capacity of the system is a normal 169 Mw (thermal) with a maximum capability of 245 Mw (thermal), which are equal to 50 and 72.5 Mw electrical, respectively. Construction of the heat sink has started and is scheduled for completion in the fall of 1962 in order that this equipment may be operated for test purposes using steam from Core 1 Seed 3 operation.

This schedule was made in order to reduce the total number of construction activities associated with Core 2 modifications.

Reconditioning and Modification of the Reactor Coolant Pump Volute

A reactor coolant volute and pump were modified in anticipation of the installation of a larger capacity pump which would be necessary for increased main coolant requirements for Core 2. Modification to the volute required internal boring; it was considered highly desirable to perform this work using a portable machine with the volutes in place. Prior to the reinstallation of this volute, a portable rig was used for the machining operation and yielded valuable experience for future in-place machining of the remaining three main coolant pump volutes, which subsequently will require similar modifications. Replacement of the modified volute during Seed 2 - Seed 3 refueling together with the portable boring rig experience is expected to facilitate the modification schedule for Core 2. The spare pump was modified for operation during Seed 3 to achieve compatibility with the Core 1 capacity pump installations.

Modifications to the volute and pump were as follows:

1. The internal bore of the volute was machined to a larger internal diameter to accommodate the larger impellers of new pumps which will subsequently be installed for Core 2.

2. The spare original Core 1 capacity type pump was provided with a spacer ring attached to the outer circumference of the thermal barrier to compensate for the enlarged internal bore diameter of the volute.

Additional System Experience

More experience (in addition to Core 1 Seed 1 experience) has been obtained for various reactor plant systems during operation with Seed 2. In general, these systems have proven satisfactory. Some of the problems associated with these systems during the period of Seed 1 operation have not re-occurred, indicating that the remedial action taken to solve them has been successful. Other difficulties have persisted, such as the boiler leak and pump problems. In a few instances, new problems have arisen and solutions are being sought. The discussion that follows will cover the major areas in which system experience has been gained during Seed 2 operation.

Water Hammer

1. Main coolant loop drains.

   Loop drainage to the coolant discharge and vent system flash tank caused severe physical pipe movement during early Seed 1 operations. Attempts to eliminate the movement by increased anchored support were successful, but calculations performed to determine the severity of the surges indicated the surges to be high for theoretical conditions. Installation of flow restriction orifices and throttling of the drain line stop check valves were not considered a satisfactory remedy. The surges have been minimized during Seed 2 operations by slowly pressurizing the drain line between the flash tank inlet valve and the loop hydraulically operated drain valve with system pressure by slowly opening the pressurizer manual drain valve. This drain line is common to all the loop drain lines. The loop hydraulic valve, which is opened next, has a zero pressure differential across it instead of full system pressure as was formerly the case. The flash tank inlet valve is then opened until the necessary drainage is completed. Below 500 psig plant pressure, the loop surges are small and a different procedure is used. In this procedure (below 500 psig) the drain line is pressurize by opening a hydraulic loop drain valve and kept pressurized in this manner until all scheduled draining is completed.

2. Main coolant loops

A series of dynamic pressure tests were performed on the reactor plant during the latter part of 1957. Pressure transients experienced during various pump switching operations were recorded. The reactor plant was operating at approximately 1840 psig and 180°F, with a nitrogen bubble in the pressurizer. The test evaluation indicated that the hydraulic surges resulting from pump switching operations were not serious in the reactor plant. No additional test for surges has been performed since that time. There has been no difficulty in plant operation to date that could be attributed to this type of surge.

Plant Warmup Rate

The allowable warmup rate for the reactor vessel above 180°F was increased from 70°F/hr to 200°F/hr for Seed 2 operation. Warmup rates approaching 200°F/hr have been used successfully, utilizing reactor heat at average coolant temperatures above 223°F. The increased allowable temperature differential of 400°F across the pressurizer surge line will permit plant heatup utilizing reactor heat from an average coolant temperature as low as 50°F during Seed 3 operation. In this way the overall plant warmup time is substantially reduced.

Cooldown and Temperature Control

1. Design basis.

Following a plant shutdown, the design of the reactor plant is such that plant cooldown can be accomplished by use of the turbine condenser unit. This procedure involves extracting heat with the condenser by slow rolling the turbine. Cooling to the lower temperature is attained by maintaining a vacuum in the condenser. The difficulties which have been experienced with this method of cooldown are: (1) the maintaining of a vacuum is troublesome because of leakage past the labyrinth seals, and (2) the cooldown is dependent on the availability of the site boiler for a steam supply to maintain a vacuum.

A design study was made to determine if a more satisfactory means of cooldown could be used. Two other significant problems existed at about the same time the turbine condenser cooldown method was questioned. These problems were: (1) the inability to obtain representative boiler water samples for chemical analysis during a cold wet layup of a boiler or group of boilers, and (2) the desire for a more satisfactory method of controlling reactor plant temperature during physics tests without requiring cycling of main coolant pumps or venting to the atmosphere.

A completely independent steam side cooling and recirculation system was provided. The system consists of a high pressure shell and tube heat exchanger, two circulating water pumps, and the necessary interconnecting piping, valving, and instrumentation. The heat exchanger is cooled by river water. It is designed to operate as a condenser at any operating pressure. The turbine-condenser unit is designed to condense while operating under vacuum only. Below 250°F the heat exchanger is operated as a water-to-water cooler since the use of vacuum is not required with this type of cooling.

2. Representative boiler water samples.

For obtaining boiler water samples in a cold laid-up boiler a flow path is set-up from the boiler steam drum through the heat exchanger to the pumps and returns it through the boiler blowdown lines. There is no cooling done by the heat exchanger in this operation. This portion of the system has operated satisfactorily and representative boiler water samples have been obtained.

3. Cooldown and temperature control below 250°F.

The temperature control and cooldown portion of the system below 250°F is set up in the same arrangement as in the setup for obtaining boiler water samples in a cold laid-up boiler except that heat is removed by the cooler operating as a water-to-water heat exchanger. Heat removal is controlled by throttling the flow of river water which is the cooling medium. This portion of the cooldown system has been operated satisfactorily.

4. Cooldown and temperature control above 250°F.

The temperature control and cooldown portion of the system above 250°F is designed to permit the heat exchanger to act as a condenser. The condensate is permitted to seek its own level (without instrumented water level controls) in the heat exchanger. The higher the level rises the less is the condensing tube surface exposed to the steam vented from the boiler steam drums. This reduces the cooling rate of the heat exchanger. The river water flow is not throttled. Heat removal is controlled by throttling the steam vented from the boiler steam drums. The condensate leaving the heat exchanger is subcooled by the cooling surface covered by the condensated water. The circulating water pumps return the condensated water through the boiler blowdown lines.

In order to avoid the possibility of thermally shocking the blowdown nozzles a temperature limitation (50°F between the water temperature in the boiler and the returned condensate) was imposed on the operation of the system. This limitation required throttling of the river water flow in order to control subcooling of the condensed water. Several informal attempts were made during Seed 2 operation for cooldown and temperature control above 250°F using the heat exchanger as a condenser. The heat exchanger operated satisfactorily but the returned condensate temperature could not be maintained within 50°F of the boiler water temperature. Present plans are to add new piping to route the condensate into the boiler steam drums to minimize thermal shock. This work will be accomplished prior to Core 2 installation.

Valve Drifts

During the early part of plant operation, considerable difficulty was experienced with valve drifts. Some hydraulically operated valves (receiving their hydraulic actuation from the valve operating system) would drift from their operating position. Often, one or more of these valves drifted (or bounced) from their positions during operation of similar valves. At other times, valves would drift from their positions without any operator action which affected the system. Also, valves would sometimes refuse to remain in the position to which they had just been moved and would drift from that position shortly after the operation was completed. These spurious motions were undesirable since they caused numerous alarms and even scrammed the reactor. The remedial action taken was to revise the operating procedure for hydraulically pressurizing valves about to be moved to a new position. The former procedure lined up the valve hydraulic cylinders in such a way that it received full hydraulic shock from the valve operating system air loading water flask. The new procedure first loads the water flask and then supplies the resultant pressure to the hydraulic valves. No actual valve drifts have occurred since the new procedure has been in effect.

Resin Discharge

The coolant purification system demineralizers serve to maintain the purity of the reactor coolant that is circulated through the reactor core. The resin in these demineralizers is discharged to the resin storage tanks in the radioactive waste disposal system when the decontamination value is reduced too low to be considered effective. The resin discharge path is a torturous run of 3-inch piping which rises some 30 feet and has many bends. Some concern was felt that the resin might not be fluidized enough during discharge to enable the demineralizers to be emptied. A successful test was performed during precritical checkout of the system to confirm that resin could be discharged. This resin, however, had not seen radioactive service and the effect of radiation on resin transportation remained to be determined. During Seed 2 operation radioactive resin was successfully discharged from both demineralizers.

Reactor Plant Component Cooling Water System

1. Cooling capacity.

   During the last three years of operation of the reactor plant component cooling water system, certain system deficiencies were discovered and corrected by modification of the system components or by revision of the operating procedures. In addition, the lowering of the reactor plant operating temperature and pressure in the summer of 1958 from 2000 psia and 523°F to 1800 psia and 500°F increased the cooling requirements of the system. The increased cooling became necessary because the main coolant pump power requirements increased due to the greater density of the reactor coolant being pumped at the lower temperature. Experience has shown that it has been necessary to reduce the station generated load during periods of high river water temperature. Reduction of station load increases the temperature (T_c) of the coolant at the main coolant pumps, thus reducing the pump power requirements and the component cooling systems heat removal load. The high river water temperature reduces the cooling capacity of the component cooling system because the river water is that system's heat sink. Station generated load is reduced until pumping temperature (T_c)

versus inlet component cooling water temperature is within the limitations specified by the main coolant pump manufacturers to prevent electrical insulation and bearing damage.

Marginal operation of the component cooling system requiring station load reduction would usually occur in early September and persist for approximately 10 days to 2 weeks. Several possible solutions were considered, namely:

a. Increase the component cooler heat transfer surface;
b. Increase the river water flow rate;
c. Install an auxiliary means of cooling;
d. Modify the main coolant pumps to increase their heat dissipation ability;
e. Replace the main coolant pumps with less restrictive ones.

No action was taken to implement any of the above possible solutions due to the scheduled replacement of the 1D main coolant pump during the Seed 2 - Seed 3 refueling period.

For a plant average coolant temperature of 500°F (486°F T_c, 3-loop operation) the pump installed in 1D loop is more restrictive than the pumps of different manufacture installed in 1A, 1B, and 1C loops. Replacement of the 1D main coolant pump with a pump of the same design as the pumps in the other loops would permit the plant to operate at full station load during periods of high river water temperature. For reasons other than the necessity to reduce station load as discussed above, the 1D main coolant pump is to be replaced during Seed 2 - Seed 3 refueling, thus eliminating the difficulty of station operation during periods of high river water temperature.

2. Temperature control.

In addition to the component cooling water system difficulty discussed above, it became apparent during Seed 1 operation that certain components could become damaged by condensation. Condensation could occur if these components become too cool with respect to the relative humidity in the plant containers. The component cooling water system is capable of cooling these components enough to cause condensation. Consequently, a limitation of 90°F minimum temperature has been standardized for all components. In certain cases this standard temperature could be reduced. The reduction would be small; however, if more than one minimum temperature were used, the station operating procedures would be more complicated than necessary for the slight benefit to be gained. To simplify control of the component cooling water system temperature a small bypass line containing a throttle valve was added to the reactor plant component cooling water system on the river water side of the system's component cooler.

Relief Valve Loop Seals

Loop seals to condense and trap pressurizer steam upstream of the pressurizer self-actuated and pilot-operated valves, and thus provide a water seal at the valve inlets, were added during the Seed 1 - Seed 2 refueling period to reduce the loss of hydrogen gas through these relief valves. The original design was based on a relief valve leak rate of 1 gph. Actual leak rates for the self-actuated valves have varied between 10 and 20 gph. Further work at increasing the effectiveness of water sealing for a leak rate of more than 2 gph would be too costly. The effectiveness will be increased to 2 gph by removal of the relief valve piping insulation. The water seal is expected to be effective following the next maintenance of these relief valves. The water seal for the pressurizer pilot operated relief valve is effective.

Boiler Blowdown

Experience has shown that the design rate originally established for sludge control (5000 lb/hr) is not effective. Test results show that the maximum blowdown rate obtainable at 500°F T_{avg} is 29,500 lb/hr. The 5000 lb/hr rate was not increased to a high blowdown rate during Core 1 Seed 2 operation because a higher blowdown rate during the short operating time left on Seed 2 would not be significantly beneficial. The boiler internals are scheduled to be inspected during the Seed 2 - Seed 3 refueling to observe the extent of the boiler water sludge deposits. A recommendation as to the blowdown rate for Seed 3 operation will be made after the boiler inspection. The 5000 lb/hr orifices were temporarily bypassed to determine the maximum blowdown rate obtainable.

Component Experience

In addition to normal maintenance of components common to all reactor plant experience there were several areas of special interest, which are covered in the following paragraphs.

Main Coolant Pumps

1. Pump failure.

The 1D main coolant pump failed to start on fast speed in the 1D loop on July 19, 1960, during a test of the reactor plant flow coastdown characteristics. During the test the reactor main coolant pumps were stopped and started in different sequences and various loop combinations. Subsequent attempts to operate the pump were met with erratic success in both hot and cold plant conditions. Investigation of the pump and controls indicated that the pump, wiring, controls, and relays were in acceptable operating condition. The pump was returned to operation. Additional testing of pump current and voltage characteristics during starting periods was conducted to determine the adequacy of the power source. A subsequent evaluation indicated that the power supply was satisfactory and that the starting difficulties were associated with pump design. This pump has subsequently been replaced with a pump similar to the pumps in the 1A, 1B, and 1C loops.

Steam Generators

1. Leakage.

The calculated heat exchanger leak rates, as indicated by I^{133} concentration in the secondary coolant, (Table IV-A) for the Core 1 Seed 2 period of operation have remained fairly constant throughout the period. One exception, however, was the 1A heat exchanger unit which developed its initial leakage in November 1960. Since its initial development, however, the leakage has remained fairly constant until May 1961, at which time the leakage increased by a factor of six for no apparent reason. Air pressure tests and a probolog survey were conducted on the 1A unit immediately following the initial detection of the leakage (during the period December 1960 through January 1961) in an effort to determine the cause and location of the leakage. The results of the air pressurization tests indicated that no visible or measurable leakage existed in the unit. Results of the probolog test indicated that of the 203 tubes surveyed, no tube was found to be cracked to a depth of 100 percent of the tube wall thickness.

Based on the results obtained from the above tests, the leak rates calculated prior to the performance of these tests, and the fact that the calculated leak rates of all four units have not increased progressively during the Core 1 Seed 2 period of operation, air testing of the four heat exchanger units will not be undertaken during the forthcoming Seed 2 - Seed 3 refueling. Consideration was given to the performance of a probolog survey and unultrasonic test on the 1A heat exchanger unit on the basis that this undertaking would provide repetitive data over an extended period of time, and that this data would be useful in connection with making a decision concerning the replacement of the 1A and 1D heat exchangers. However, since the data gained as a result of this test could not be obtained and evaluated in sufficient time to be of use, the probolog and ultrasonic tests were not undertaken.

TABLE IV-A

PRIMARY TO SECONDARY HEAT EXCHANGER LEAK RATES*

(ml/min)

Month	1A	1B	1C	1D
May 1960		1.20	0.47	†
June		0.64	0.24	†
July		0.64‡	0.24‡	†
August		0.69	0.27	†
September		0.69	0.41	†
October		0.69	0.41	†
November	0.61	0.82	0.52	†
December	0.20**	0.50	0.40‡	†
January 1961	----	0.50‡	0.40‡	†
February	0.40‡	0.60‡	0.50‡	†
March	0.30‡	0.40‡	0.40‡	†
April	0.80	0.39	0.28	†
May	2.5 ††	0.35	----	----
June	2.7 ††	0.20	----	----
July	2.6	----	----	----

* Leak rates are based on I^{133} activity in the primary and secondary systems.

† Not detectable.

‡ Leak rate based on gross iodine (actual calculation not performed).

** Leakage just prior to conductance of an air leak test.

†† 1A boiler in hot layup condition.

2. 1A and 1D steam generators.

Performance tests on the 1A steam generator indicated that a low flow condition existed in the first downcomer at the heat exchanger inlet. A test was run with varying water levels, with the station operating at full power (on 3 loops) in order to determine the relationship between drum water level and the low flow condition. The tests showed that flow reduction occurred at drum levels above 3-1/2 inches over normal drum level.

Two steam probes were installed on the steam generator to check for steam blanketing of the heat exchanger tubes due to the low flow condition. Conductivity samples were obtained with the probes. It was concluded that steam blanketing did not exist. In conjunction with this test, a partial performance test was made to provide a correlation between conductivity and performance. It was concluded, also, that chemical hideout was not occurring.

The low flow condition in the 1A and 1D boilers was corrected by seal welding between the downcomers and the steam drum wrapper sheet and installing vortex breakers at the downcomers inlets.

Valve Operating System

1. Failure of 3-way valves.

Several 3-way valves in this system have had their wave spring and seals fail. The failure permitted valve leakage. The springs serve initially to hold shear type seals against a flat highly polished stellite surface on a cylindrical valve disc. Unbalanced pressure forces across the seals press the seals and discs together to effect a tight seal. Parts of the springs have entered the primary system piping and present the possibility of interfering with other valve operations. A new spring design using stainless steel rather than beryllium copper springs material has been installed in these valves. A cover skirt design will be incorporated in the design to prevent broken spring parts from entering the primary system piping.

2. Modification.

The air filter and moisture separator of both the normal in-service air compressor and its spare unit are to be replaced in order to improve the safety of the units against oil vapor hazards. In addition, an investigation concluded that it would be too costly to modify the spare unit as an automatic control operating in parallel with the normal in-service unit.

Pressurizer Vessel Heater Wells

Six pressurizer vessel heater wells were removed during Seed 1 - Seed 2 refueling. An evaluation of these heater wells indicated that, after 10,500 hours of plant operation, boiling of coolant in the crevice areas had resulted in no evidence of stress corrosion.

Reactor Plant Equipment Layup

New layup techniques are being incorporated into the layup procedures in order to provide better protection for operating equipment that is temporarily not in use. Special attention has been given to layup of main coolant pumps to avoid their contact with oxygen by use of a nitrogen blanket during and after pump draining.

Strainer Replacement

The basket type strainers in the reactor plant are being replaced with knife-edge strainers to reduce the possibility of strainer rupture, allowing failed parts to enter the primary system.

Flash Tank

The coolant discharge and vent system flash tank is a carbon steel vessel. Internal tank corrosion resulted in a program to find a suitable surface coating material. Testing of several possible coatings resulted in a negative evaluation. The investigation is being continued by Bettis plant modification for Core 2.

Relief Valves

The pressurizer vessel self-actuated steam relief valves and the reactor vessel self-actuated water relief valves have developed excessive leakage during Seed 2 operation. The leak rate has increased with in-service time. The valves will be repaired during Seed 2 - Seed 3 refueling.

In an effort to provide a means of determining relief valve leakage, a collection system was installed. This arrangement consisted of a sample cooler and collection vessel with a sightglass for visual level indication.

Electrical System Experience and Modifications

Nuclear Instrumentation System

The principal problems experienced with the nuclear instrumentation system centered about the coaxial cable connections at the neutron detectors in the neutron shield tank. The connector problems were as follows:

1. Irradiation damage to the insulating inserts in the connectors, first experienced during Seed 1 operation. This problem has been minimized during operation of Seed 2 by replacing all teflon inserts, which are highly susceptible to irradiation damage, with inserts of rexolite, a cross-linked polystyrene.

2. Irradiation and consequent activation of the tinned brass connector bodies, also experienced since initial station operation. While this problem has not impaired the performance of the system, it has resulted in an undesirable amount of radiation exposure to maintenance personnel handling detectors. All connectors near the reactor are to be replaced with units having aluminum bodies at the conclusion of Seed 2 operation.

3. Mechanical damage to cables at the detector wells. The flexing and handling of cables when removing and inserting detectors in the detector wells has resulted in substantial physical deterioration of the cable sections at the detector ends. In addition, many cases of insulation failure at the end connector have resulted from cable movement. It is intended to replace all cable sections adjacent to the detectors during the Seed 2 - Seed 3 refueling period. Since the existing cables are continuous from the detector connections to the reactor plant container electrical penetrations, it will be necessary to install junction boxes on the reactor walkway in order to effect the replacement. The installation of the junction boxes will permit periodic replacement of the cable sections subject to handling in the future.

Performance of the neutron detectors themselves has been satisfactory during Seed 2 operation. No replacement of any of the four compensated ion chambers used for intermediate and power range instrumentation has been necessary. One source range BF_3 element was replaced during Seed 2 operation due to a failed connector. The large number of BF_3 failures experienced during Seed 1 operation prompted the tube manufacturer to institute production changes which resulted in no BF_3 failures during Seed 2 operation.

The most significant modification to the nuclear instrumentation system made since initial station operation was carried out during the Seed 1 - Seed 2 refueling period. This modification encompassed the permanent installation of a fourth channel of source, intermediate, and power range equipment. The original installation consisted of three channels of such equipment; the need for a fourth channel became apparent early in Seed 1 operation when it was discovered that the core was subject to spatial power oscillations, and that, as a result, all four core quadrants required monitoring and protection. A fourth channel was added on a temporary basis to provide four-corner protection during the latter part of Seed 1 operation, and the permanent installation was delayed until Seed 1 was depleted. The permanent incorporation of the fourth channel included modification of the readout on the reactor control console. The relatively small instruments on the console, which presented the readouts of all channels simultaneously, were replaced with larger, more readable switchboard-type instruments and associated selector switches to permit selective readout of the individual channels. To compensate for the loss of simultaneous indication to the operator, auctioneering circuitry was added for source range log level and start-up rate, intermediate range log level and start-up rate, and power range linear level. The selector switches associated with the new meters contain auctioneer position, in which the highest of the four channel outputs is presented to the operator. The auctioneering mode of operation is normally used.

It was noted late in Seed 2 life that there was interaction between the four power range channels. An investigation disclosed that the cause was a decrease in the back resistance of the diodes in the auctioneering circuitry. Replacement of the diodes corrected the situation.

Other minor modifications to the nuclear instrumentation system included:

1. Installation of knobs on alignment potentiometers in the nuclear equipment to eliminate the use of screwdrivers.

2. Installation of glass panels in the doors of the nuclear equipment panels so that meter indications and switch positions could be readily observed.

3. Modification of the compensated ion chamber microammeter shunt arrangement to eliminate small errors introduced when changing power range amplifier input taps.

4. Addition of terminal blocks on the power range amplifiers to facilitate changing of input taps. This operation previously required resoldering of magnetic amplifier connections.

Remote Viewing System

The performance of the closed circuit television systems used for remote monitoring of the steam drum sightglasses and for general area viewing within the reactor plant container has not been satisfactory. Even after earlier equipment modifications by the manufacturer, an excessive amount of maintenance effort has been required. All cameras and monitors are to be returned to the manufacturer during the Seed 2 - Seed 3 refueling period for additional reworking. It is believed that this action will result in improved stability of internal circuitry, which in turn will make selection of electronic tubes less critical and increase the time interval between alignments.

Reactor Rod Control System

The only modifications made to the rod control system consisted of extending the safety shutdown tripping function to include the variable voltage power supplies to the spare inverters, and electrically interlocking all in-hold-out control switches so that only one switch could produce rod motion at a given time.

Reactor Power and Temperature Control System

The reactor power and temperature control system remained out of service throughout Seed 2 operation. The installation of a low T_{avg} safety shutdown setpoint in 1958 to protect against rod drop accidents has prevented the use of the automatic control system, because the system would tend to cancel out the reduction in average coolant temperature accompanying a dropped rod. The system outputs to the rod control system were physically disabled to eliminate the possibility of the reactor being accidently placed in automatic control. No difficulty was experienced in controlling the reactor manually.

Failed Element Detection and Location System

While the station was operating with Seed 1 in service, the FEDAL system indicated a failed fuel element at blanket position F-2. This blanket element was replaced during the Seed 1 - Seed 2 refueling period. However, the FEDAL system continued to indicate a failed blanket element at position F-2 when the station was returned to power. In attempting to determine whether this condition was indicative of a defective replacement fuel element or an erroneous indication from the FEDAL system, it was postulated that the multiport valve might be rotating in the wrong direction, thereby causing the presumed sampling sequence to be incorrect. The direction of rotation of the multiport valve, a hermetically sealed unit, was shown to be incorrect by suspending a small bar magnet adjacent to the stator winding and observing the rotation direction of the magnet as the multiport valve revolved. The sampling sequence was then redesignated to agree with the actual movement of the valve.

It was noted that the sampling cycles frequently contained one less sampling position than the physical number of such positions in the multiport valve, indicating that port skipping was occurring. The cause of the skipping was found to be faulty resetting of the valve, whereby the valve was occasionally starting the sampling sequence at port 2 instead of at port 1. It is intended to correct this condition by a modification to the electrical control circuitry.

Difficulties were also encountered with the high-voltage power supply in one of the two sample activity monitors, as evidenced by repeated failures of the high-voltage transformer. The other monitor, which is of identical design, was not subject to such failures. The power supply of the defective monitor has been completely rebuilt in an effort to eliminate the difficulty. Consideration is also being given to redesigning the high-voltage power supplies in order to eliminate the special components utilized in the present design.

The original readout instrumentation utilized separate flow and activity recorders for each of the two monitoring trains. These recorders have been replaced with a single two-pen recorder for each train. The presentation of both sample flow and activity on a single chart permits easier correlation of individual activities with the sampling positions.

Primary Plant Control System

The following modifications to the primary plant control system were made or will be made as a result of Seed 2 operating experience:

1. Improvement of ventilation in the panel housing the hydraulic valve position indicator receivers to eliminate overheating of components and minimize drifting of setpoints.

2. Modification of hydraulic valve position indicator receiver test circuitry to permit calibration of receivers while the station is operating.

3. Addition of a position to the pressurizer heaters master control switch on the control console to permit the reactor operator to energize all heaters from the console under abnormal low-pressure conditions.

4. Addition of audible alarms to reactor plant container airlock doors to alert a person attempting to enter an airlock if the companion door is not secured. Electrical interlocks prevent the actual opening of a door under this condition.

Primary Plant Instrumentation System

The following modifications were made to the primary plant instrumentation during Seed 2 operation:

1. The temperature interlock which permitted placing an isolated loop in service with the reactor critical was removed. In its place was substituted interlocking which prevented returning a loop to service unless all control rods were bottomed. The reasons for revising the interlocking were, first, the permissible temperature difference between the incoming loop and the remainder of the system, at which a loop could be cut in, had narrowed with increased

core life to a value which the original interlock could not distinguish, and second, the operating advantage of the interlock was outweighed by the maintenance attention which it required. Since the loading on a utility system follows a predictable cycle, it is not an undue inconvenience to schedule a station shutdown during a light load period for the purpose of bringing in a down loop.

2. The scanning rate of the automatic temperature scanners was reduced from four points per second to forty points per minute. The purpose of the decrease in scanning rate was to prolong the life of the scanning switches used in the equipment. In addition, there appeared to be some lack of correlation between the temperature indications obtained from similar points. Therefore, all thermocouple circuits are to be "padded" in such a fashion that equal resistances will be obtained and all indications can be adjusted from a standard base.

The electrical null-balance instrumentation used in measuring reactor plant flows, pressures, temperatures, and levels has performed satisfactorily. The amount of drift experienced between calibrations has been small. The instrumentation is calibrated every six months in order to maintain a high degree of accuracy in test data taken from the station.

Several failures of differential pressure cells have occurred, primarily bellows ruptures due to overpressure or failure of the internal relief valves to reset. Precautions have been taken to insure that the correct valving sequence is followed when removing or returning a cell to service, as cell failures were generally the result of incorrect valving sequence.

After the end of full power capability of Seed 2, the reactor was operated at an average coolant temperature of 475°F to accumulate additional operating time on the core. The average temperature instrument was re-ranged to satisfy these operating conditions.

Radiation Monitoring Equipment

Data evaluated from Seed 1 - Seed 2 operation has resulted in the deletion of the following items as unnecessary:

1. The valve operating cubicle monitors
2. Resin discharge monitors on main discharge line
3. Resin storage and surge tank monitors

The following items have been repositioned or rearranged either to obtain better operation or to be in line with the service building modifications:

1. Boiler compartment monitors
2. "Friskers"
3. Hand and feet counters
4. Exit monitors

The following items were added as a result of actions recommended by the emergency plan committee:

1. Two rooftop monitors on the fuel handling building
2. A high level effluent water monitor in waste disposal

Equipment difficulties have been experienced as follows:

1. It was found that all channels which are recorded required potentiometer adjustments to cause the channel meters and recorders to agree.

2. The air particle detectors have indicated a high failure rate. A new spare detector was installed in channel 12 of ORMS. The replaced unit will be reworked as a spare.

3. The stack gas monitor in waste disposal was found to be less sensitive than expected. A new unit is being installed with a sensitivity of 10^{-6} uc/cc.

4. The effluent water monitor has been removed from service and is being tested in the Bettis laboratory as a result of the calibration difficulties experienced during Seed 2 operation.

Data Acquisition System

Electronic equipment has been installed which will provide a printout of some 145 selected signals related to plant operating conditions. This system will be used to obtain test data and is not required for ordinary reactor operation. The equipment is located in an area separated from the control room. A digital computer continuously scans the various reactor and turbine plant parameters, such as pressure, temperature, flow, flux level, control rod positions, etc. The information is stored in memory until the program demands that the data be logged out by automatic typewriters. This information may also be put on punched cards by use of the equipment memory circuits that feed information to a punch machine. The punch cards will be made available for use in the Bettis computers where further evaluation of test data can be made. In addition, a physics console has been installed in this same area. This console contains the scalers and associated printers which receive signals from the detectors used for core physics testing. This relocation of instruments relieves the congestion in the control room area.

CHAPTER 4

REACTOR PROTECTION SYSTEM PERFORMANCE

Introduction

PWR Core 1 Seed 2 has provided a year and a half of additional operating experience with the Shippingport Reactor Protection System. Whereas Seed 1 operation provided information of a general nature, Seed 2 operation has provided additional detailed data for evaluating a protection system which consists primarily of magnetic amplifiers. Experiences encountered in time response measurements, set point drift characteristics, and system interaction are detailed. The following presents a detailed review of the experience gained with the reactor protection system during Seed 2.

General

Following Seed 1 a review was made of the operational history of the reactor protection system to determine whether system simplifications were possible which could result in greater system reliability. The review was made by reexamining the present system, considering changes to the system and studying the Seed 1 incident reports and the component failure records. The following points were determined:

1. The system provided dependable protection during Seed 1.
2. Maintenance or component failure was not excessive.
3. The component problems which took place were confined to the bistable magnetic amplifiers (BMA's).
4. The installed system was complex and provided flexibility which was not used during Seed 1 operation.
5. Precritical adjustment and alignment required considerable operator action.

In evaluating the points listed above versus system changes which could be effected by Seed 2 operation, it was concluded that any design changes considered would have to be limited to a rearrangement of the present components. Available time precluded the procurement of major components. It was concluded that there was not sufficient justification to change the present system for Seed 2 and that more benefit could be gained by investigating the combined systems, nuclear instrumentation, and reactor protection to determine what design changes should be made for future seeds or cores.

Seed 2 Reactor Protection System Set Points

From Reference 1, it can be seen that the power limits of Core 1 Seed 1 were determined at various times during Seed 1 operation. The redetermination of the power limits was done to take into account:

1. The improved nuclear design data
2. The results of experimental work in heat transfer
3. Design parameters as measured from the operating core
4. Performance of instrumentation and controls in maintaining power limits.

Set points were determined for the following times during Seed 1 life:

1. 0 to 3000 EFPH
2. 3000 to 5000 EFPH
3. 5000 EFPH to the end of Seed 1 life.

The determination of Seed 2 set points followed a similar pattern to that established during Seed 1. The criteria used in determining the set points were:

1. Departure from nucleate boiling ratio (DNB) in the seed or blanket should be equal to or greater than 1.5 under equilibrium conditions.

2. A surface heat flux of 500,000 BTU/hr-ft^2 in the blanket material should not be exceeded.

3. No bulk boiling in steady-state operation.

Table IV-B presents the Seed 2 power versus flow settings.

TABLE IV-B

POWER VS FLOW PROTECTION SET POINTS

	Power Versus Flow Settings Until Group III Control Rods Reach 27 Inches		Power Versus Flow Settings When Group III Control Rods Exceed 27 Inches		Power Versus Flow Settings After the End of Full Power Capability	
	Seed 2		Seed 2		Seed 2	
Flow Conditions	Insertion	Scram	Insertion	Scram	Insertion	Scram
4 or 3 loops, fast speed	114	118	111	114	90	95
4 or 3 loops, slow speed	24	36	24	36	24	36
2 loops either speed	--	24	--	24	--	24
Less than 2 loops, either speed	--	0	--	0	--	0
All start-up and cold plant conditions, either speed, 2 or more loops	--	24	--	24	--	4

Note: 100% power - nominal 231 Mw(t)

Seed 2 Operating Difficulties

Seed 2 operation disclosed several areas of apparent protection system difficulties. These areas were:

1. Excessively long response time of excess neutron level signal.

2. Drift in set points.

3. Firing of memory magnetic amplifiers without the actuation of the corresponding protection circuitry.

4. Interaction between bistable magnetic amplifiers (BMA's).

5. Difference in set point of the BMA's when using the normal system relays versus the test set relays.

6. Difficulty in aligning and checking the 3N and 4N series BMA's.

The succeeding paragraphs discuss the above considerations in more detail.

Excessively Long Response Time of Excess Neutron Level Signal

When the results of the second performance of test procedure DLCS 12901, reactor protection system, were reviewed in April 1960, the scram response times of the excess neutron level signal were found to be as high as 2.2 seconds whereas the results of the first performance in November 1957 indicated 0.4 seconds. This indicated a gross change in system time response.

It was determined by laboratory tests that the method of testing the response time of the circuitry was in error. DLCS 12901 specified the use of a neutron level step input signal of 1 percentage point over the trip point (1 percent overshoot). The system was checked with a neutron level signal which exceeds the trip point by 10 percent and proper response times were obtained. Laboratory investigation determined that a 10 percent overshoot of the neutron level set point produced time responses that were conservatively equivalent to approaching the trip point at the worse case linear rate of nuclear level change of 100 percent/sec. A review of the fast-speed recorder charts obtained during the first performance of DLCS 12901 in November 1957 revealed that the specified method of applying a step input signal of 1 percent overshoot was not followed.

It was determined that step inputs considerably greater than 1 percent were applied to obtain the time responses tabulated in the test results. Therefore, it was concluded that the long time responses were always present for 1 percent overshoot signals and that no gross change had taken place since November 1957.

Drift in Set Points

Shippingport operating experience has indicated that the bistable magnetic amplifiers (BMA's) appear to have an undesirable drift characteristic, particularly when the temperature of the relay room in which the BMA's are located varies. This drift results in a corresponding drift of the set point of the bistable.

Four spare bistables were tested in the electrical laboratory at Bettis. The circuit configuration used approached the actual circuit as installed at the site. The control winding of the four bistables and a 819-ohm resistor (to simulate the control winding impendance of a fifth unit) were wired in series and connected to the output of the nuclear instrument system (NIS) power range amplifier. In actual system operation the control windings of five bistables are wired in series. The output of each of the bistables was wired to a control winding of a linear magnetic amplifier.

The bistables were aligned to pick up in sequence with approximately 0.2 milliamperes (MA) difference between pickup points of successive bistables. This was necessary to minimize interaction between units. The input to the NIS power range amplifier was varied such that the output of the NIS amplifier would increase in a ramp from about 5.2 MA to 6.2 MA at a rate of 0.016 MA per second. This ramp increase was continuously repeated and automatically controlled. The bistables were placed in a thermostatically controlled oven and the cycling input to the bistables performed at various temperature levels from 90°F (ambient) to 145°F for from 4 to 12 hours at each temperature level. The input current to the bistables and the output voltage of each bistable was continuously recorded. Samplings of the data are presented in Figure IV-12. The data indicated variation in trip point with temperature ranging from 0.1 MA to 0.15 MA, which corresponds to 2 to 3 percent of full power (FP). It was noted that the trip point of BMA's No. 1 and 2 decreased with increasing temperature, whereas the trip point of BMA No. 3 increased with temperature. The feedback of unit No. 3 was modified prior to the test in an attempt to obtain better bistable response. The increase of set point with temperature is not desirable since reactor protection is reduced. The data also indicated that the repeatability of the trip point at a given constant temperature ranges from 0.025 MA to 0.09 MA, corresponding to 0.5 to 1.8 percent FP (Figure IV-12). This is comparable to the 2 percent variation of trip point with temperature over the expected temperature range of the relay room (85°F to 105°F).

It was concluded from this test that the trip point of the BMA's may vary by as much as 2 to 3 percent FP when subjected to temperature variations of 90°F to 145°F, and by as much as 2 percent FP over the expected temperature range of the relay room. The repeatability of checking the BMA trip points by manually adjusting the test current was demonstrated in the Laboratory and is illustrated in Table IV-C. If care is not exercised in setting or checking the trip point of the BMA's, the repeatability of the trip point may exceed 2 percent FP, which is comparable to the temperature drift. Changes to the operating procedures were recommended and specified, that as the test current approaches the anticipated trip point, the test current should be increased in small increments (about 1 percent) and held for 10 seconds before increasing the test current further. Experience has shown that this has improved the repeatibility of the trip point.

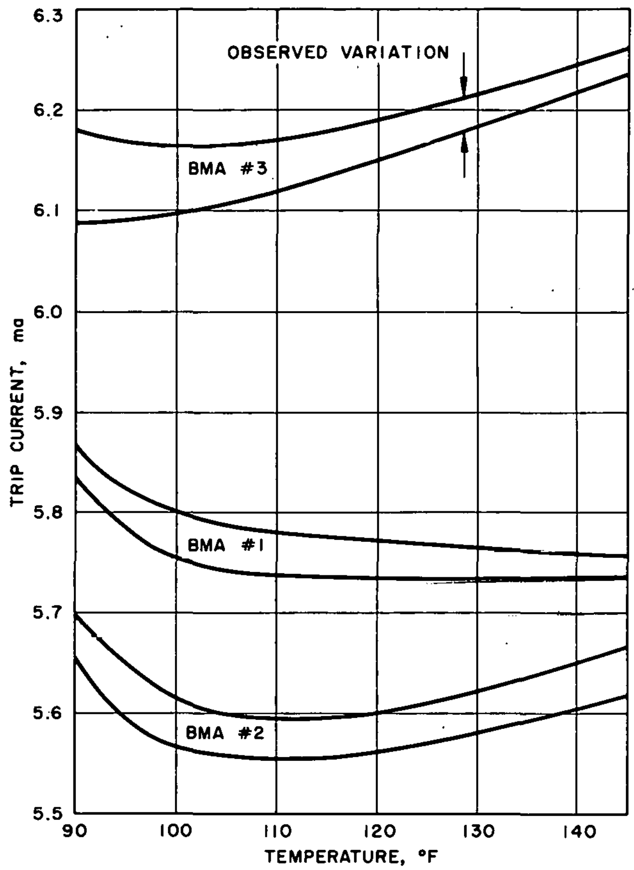


Figure IV-12. Variation of BMA Trip Point vs Temperature.

TABLE IV-C

VARIATION OF BMA TRIP POINTS

Temperature (°F)	BMA	Trip Point (MA)	Drop Out (MA)	Output (volts)
95	#1	5.44	5.02	4.91
		5.42	5.0	4.9
		5.40	4.97	4.85
95	#2	5.64	5.32	4.9
		5.47	5.07	4.8
		5.47	5.07	4.75
95	#3	5.72	5.32	5.2
		5.82	5.27	5.25
		5.7	5.27	5.2
95	#4	5.67	5.13	5.15
		5.67	5.13	5.12
		5.65	5.12	5.14

Firing of Memory Magnetic Amplifiers Without the Actuation of the Corresponding Protection Circuitry

An incident occurred during Seed 2 in which a memory magnetic amplifier (MA 25) fired without the actuation of the corresponding protection circuitry. MA 25 is the memory unit for the safety insertion circuitry, and, as such, it should not be energized unless an insertion has taken place. The input to the safety insertion circuitry is provided by the 4N series BMA's. Tests at Shippingport indicated that while the 4N series BMA's exhibited satisfactory bistable action, the 4N2 unit required readjustment. The initial testing of this unit revealed a slight oscillation in the BMA output voltage when the input was within 1 percent of the trip point. This oscillation caused the memory magamp to fire without actuating the safety insertion amplifier. The load resistance of the 4N2 unit was increased slightly to improve the bistable action and minimize the oscillation in the output. Satisfactory operation was obtained.

Interaction Between Bistable Magnetic Amplifiers (BMA's)

As a part of the test described above, the bistables were further tested to determine their interaction effects. It was desired to determine the effect on the trip point of connecting BMA's in series and varying the setpoint of one of the BMA's and observing the change in setpoints of the other BMA's.

The effect of connecting BMA's in series was accomplished by aligning the BMA's and then replacing three units, one at a time, with a 819-ohm resistor. The results clearly indicated that the pickup point is affected by the number of BMA's connected in series. The interaction of set points which results from varying the set point of one of the BMA's and observing the change in set points of the other units was clearly demonstrated.

From this test it was concluded that the BMA's should be aligned in order, low trip point set first to minimize interaction effects. If the set point of a BMA is to be adjusted a significant amount, the set point of all BMA's must be checked to ensure that the desired alignment is achieved. Connecting from 1 to 4 BMA's in series may cause the trip point to vary as much as 0.9 MA. This effect must be considered when testing the BMA's out of their normal circuit arrangement.

Different in Firing Point of the Bistable Magnetic Amplifiers (BMA's) When Using the Normal System Relays versus the Test Relays

Seed 2 checks of the reactor protection system indicated that the trip point of the 4N series BMA's varied by as much as 5 percent depending on whether the normally installed or test relays were connected to the safety insertion amplifiers. It was believed that this discrepancy results from the BMA's behaving linearly rather than as bistables. It was subsequently determined by testing that the variation of the trip point was not dependent of the relays but on the status of the 5N series BMA's (i.e., trip point set for cold plant protection or higher power to flow scram). Table IV-D illustrates the 4N series BMA trip point dependency on the status of the 5N series units. A change in alignment and checking procedure of the BMA's was provided to the site and specified that the BMA's be aligned in sequence, lowest trip point first, with the 5N series BMA's biased for excessive nuclear level and not for cold plant protection.

TABLE IV-D

EFFECT OF RELAY AND STATUS OF 5N BMA's ON 4N BMA's

BMA #1	Relay (S.I. or Test)	5N Set Point (%)	BMA Trip (%)	BMA Output (volts)
4N1	Test	24	83	3.82
	S.I.	*24	83	3.82
	Test	95	91	4.1
	S.I.	†95	92	4.08
4N2	Test	24	85	5.4
	S.I.	24	86	5.5
	Test	95	90	5.8
	S.I.	95	89	5.8
4N3	Test	24	86	4.2
	S.I.	24	86	4.35
	Test	95	91	5.4
	S.I.	95	91	5.4
4N4	Test	24	86	4.8
	S.I.	24	87	5.0
	Test	95	89	5.3
	S.I.	95	89	5.3

* Cold plant protection in

† Cold plant protection out

Difficulties in Alignment and Checking the 3N and 4N Series BMA's

Seed 2 operation indicated continuous operator difficulty in aligning and checking the 3N and 4N series BMA's as the two sets of units were both adjusted to trip at 111 percent FP. In checking a given 3N BMA it was difficult to determine whether the 3N unit being checked or the accompanying 4N BMA had tripped.

In the Seed 2 arrangement of the reactor protection system, the 3N series BMA's provide for a safety insertion at 111 percent power with 3 loops at full speed, and a safety shutdown at 111 percent with 2 pumps at full speed. The 4N series BMA's provide for a safety insertion at 111 percent power, independent of the flow condition. The 1N series BMA's provide for a safety shutdown at 24 percent power with less than 3 loops in operation.

It was, therefore, concluded that the 3N series BMA's could be disabled without compromising reactor protection. The set point of these BMA's was adjusted to a maximum value.

Performance of the System

A review of the incident reports for Seed 2 indicated that only two incidents occurred which required system action to prevent the reactor from actually experiencing conditions beyond design limi One of these incidents was a safety shutdown due to low system pressure brought on by operator erro and the second incident was a safety shutdown due to high coolant temperature. In addition, a number of insertions and shutdowns were the results of component problems and operator errors. The following presents a description of the two incidents which required protective action.

On June 23, 1960, the reactor coolant temperature was being increased at a rate of approximately 150°F/hr, using the reactor as the source of heat in preparation for a station start-up. As the indicated pressurizer temperature approached 610°F, the indicated pressurizer pressure increased to 1850 psig, and the pressurizer level rose to 190 inches. To relieve the high pressurizer level and pressure, the operator attempted to drain coolant from the system. However, due to insufficient valve operating system air pressure, the 1A loop drain valve did not respond. Alternatively, the high-pressure condition was relieved by opening the pressurizer spray valve and manually inserting the rods. A rapid pressure decrease to 1400 psig resulted in the safety shutdown.

On January 28, 1961, during a rapid station shutdown in accordance with test procedure DLCS 34401, the load was decreased from 65 Mw gross to 0 at the maximum rate available (approximately 0.7 Mw/sec) using the main generator control. The generator breaker was then opened and the turbine throttle valves were tripped. The test procedure required that the decay heat motor-operated stop valve be open and the control rods be inserted upon actuation of high T_h alarm (517°F) or upon increase of average coolant temperature to 510°F. However, due to the rapid coolant temperature increase initiated by the load reduction, the controlling rod group was manually inserted when the average coolant temperature reached 507°F. In spite of this anticipating action and the lifting of the decay heat removal relief valve, the coolant temperature continued to increase until the hot leg temperature reached 522°F and initiated a safety shutdown.

Conclusions

Seed 2 experience has shown that the Shippingport Atomic Power Station is operated in a manner such that reactor protection system action has been held to minimum. When actual core protection was required, i.e., during the low pressure and high T_h safety shutdowns, the system provided dependable protection. Both Seed 1 and Seed 2 operation has shown that shutdown set points of 114 to 118 percent are not restrictive. Five out of six of the operating difficulties described herein were resolved by changes to the testing and operating procedures rather than by changes to the equipment. It is believed that the operating difficulties which have been experienced with the system will be nominal during Seed 3 by following the modified alignment procedures.

REFERENCE

1. "Shippingport Operations; From Start-up to First Refueling December 1957 to October 1959," DLCS-364.

CHAPTER 5

REACTOR PLANT FUEL CANAL PROTECTIVE COATING

Introduction

The fuel handling canal (Figure IV-13) contains the reactor pit, fuel storage pit, deep pit, shroud storage pit, disassembly and transfer areas, crane lock, and related sumps. These pits, locks, and sumps are constructed of reinforced concrete. During power operation the reactor pit contains demineralized water and the same is true for the fuel storage pit when irradiated core components are stored. The other areas are filled with demineralized water during refueling operations and other related periods.

During refueling and fuel storage periods, this demineralized water becomes contaminated with radioactive crud. Since laboratory studies and experience have shown that concrete, once contaminated, cannot be easily freed of the radioactive contamination, it is necessary to protect all concrete surfaces in the canal with a coating. For this application the coating should be resistant to high levels of radiation, to demineralized water, to contamination, and to decontaminating solutions that may be required.

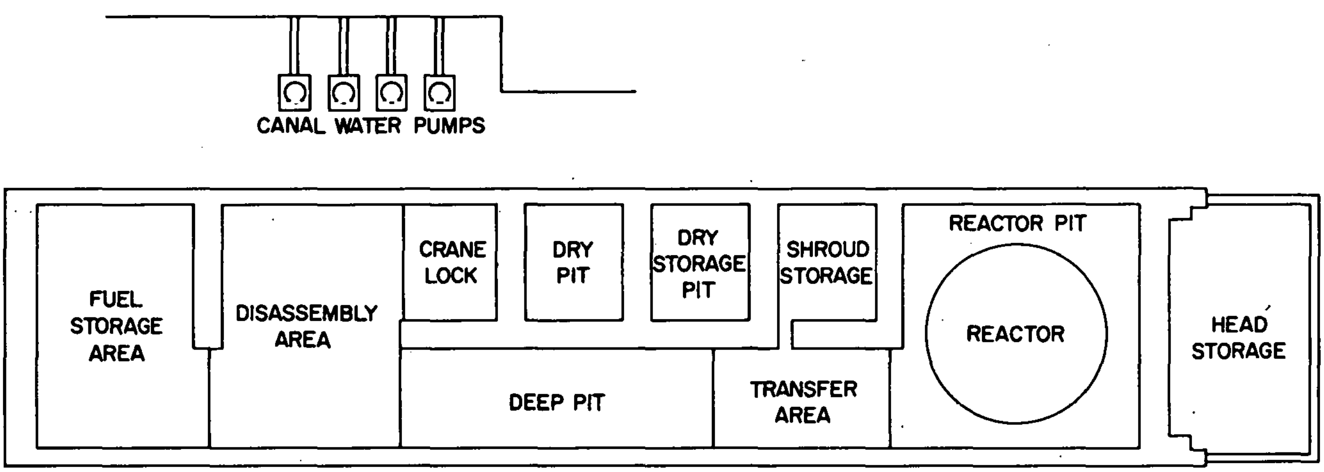


Figure IV-13. Fuel Handling Canal.

History

As construction of the Shippingport Station was approaching completion, it was decided that the various pits should be coated with a protective coating. In addition, the walkways and pits that do not contain demineralized water should also be protected against radiation build-up from airborne contamination. These applications were to be accomplished prior to initial power operation.

In 1957 a protective coating consisting of a cross-linked, thermal setting, styrenated polyester resin was applied to the concrete surfaces of the canal. The coating was selected on the basis of the manufacturer's claim that a considerable saving in time could be gained since the use of this polyester coating would eliminate the necessity of stone finishing the form-cured surfaces. The concrete pits were then etched with muriatic acid, rinsed with water, and allowed to air-dry. The following procedure, recommended by the manufacturer, was then adopted:

1. A coat of high viscosity primer was brushed on to fill the voids in the concrete surface and the excess removed with a squeegee.

2. A coat of regular primer was rolled on and allowed to cure to tackiness.

3. A layer of two ounce glass fabric was pressed into the tacky surface.

4. A saturation coat of regular primer was rolled into the glass fabric and allowed to cure to tackiness.

5. A final coat was then rolled on.

To complete the protection of the canal material from contamination, the structural steel components in the canal were protected with a polyvinyl coating. There were some failures of this coating on the canal gates which were repaired with the same material.

A few days after the initial application of the polyester, amber colored spots were noticed on the canal surfaces. These spots were readily removed by washing with water and no significance was attached to them. In 1958 an inspection of the canal revealed a number of large blisters which the original applicator repaired in October 1958. Another inspection in 1959 pointed out that the reactor pit contained numerous pinholes and small blisters from which a colorless liquid, with the appearance of water, exuded. In other areas small brown stains were observed on the outer surface of the coating. As a result of a discussion with the manufacturer, an additional coat of the polyester material was applied over the original coat as a possible remedy to the blisters and pinholes.

Immediately after the application of this additional coat of polyester material an exudation of a brown material was observed from numerous pores in the newly coated surface. This exudate was colorless when first formed and then changed to a brown liquid upon exposure to air. Similar defects were noted throughout the canal. A sample of the material and concrete was analyzed and the results are presented in Tables IV-E and IV-F. Laboratory as well as field tests showed that blisters and pinholes containing the brown exudate could not be repaired without complete removal of the polyester material to the bare concrete. Based on this evidence, and with the refueling of Seed 1 with Seed 2 rapidly approaching, it was decided to remove the coating from the affected areas to the bare concrete and to recoat.

When the applicator started to remove the blistered area, it became apparent that a very high percentage of the polyester was not bonded to the concrete. In fact, the entire coating in some areas could be stripped from the concrete surface by hand. As a result, over 90 percent of the polyester was removed from the concrete in the pits of the canal. The following procedure was adopted for application in August 1959 under the direct supervision of contractor engineers.

1. After removal of the original coating, the concrete was disc sanded.

2. All the voids in the concrete were filled with a block filler and the excess removed with a squeegee.

3. A coat of regular primer was then applied and as soon as it was tacky, a 10-ounce glass fabric was pressed on.

4. After at least four hours elapsed, a saturation coat was rolled on and squeegeed into the glass fabric. As soon as this coat was dry, a coat of surfacer was applied to fill all voids and pinholes.

5. Another coat of surfacer became the final coat.

TABLE IV-E

ANALYSIS OF CONCRETE SCRAPED FROM HEAD STORAGE PIT

Water Soluble - 7.6%	Water Insoluble - 92.4%
OH - 1.2%	Silica as SiO_2 - 58.6%
CO_3 - 0.7	Iron as Fe_2O_3 - 5.6
Cl - 0.03	Aluminum as Al_2O_3 - 5.1
SiO_2 - 1.1	Phosphorus as P_2O_5 - 0.0
Ca - 2.9	Sulfur as SO_3 - 0.0
MgO - 0.54	Calcium as CaO - 11.8
Fe - 0.002	Magnesium as MgO - 2.7
Na - 0.41	Ignition loss at 750°C - 8.4
SO_4 - 0.0	
Ignition loss at 750°C - 0.8	
pH at 27°C (0.1% Solution) - 10.6	
Sp Cond (0.1% Solution) - 200 μmhos/cm	

TABLE IV-F

ANALYSES OF EXUDATE

Loss in weight at 105°C	18.2%
Ignition loss at 750°C	96.7%
Water soluble	100%
Specific conductivity (0.1% solution)	480 μmhos/cm
pH of 0.1% solution	5.4%
Hardness as Ca	0.5%
Sodium as Na	2.0%
Iron as Fe	< 0.02%
Silica as SiO_2	< 0.02%
Chloride as Cl	< 0.20%

After the final coat was applied, considerable touch-up work was required. Many pinholes appeared that were extremely difficult to correct. In some areas, two or three additional coats were required in an attempt to eliminate this phenomenon. Prior to Seed 1 - Seed 2 refueling another inspection revealed a number of large blisters in the reactor and fuel storage pits which were repaired. Following the refueling of Seed 1 with Seed 2, the polyester coating in the reactor pit was again inspected in April 1960. Numerous cracks, blisters, and pinholes were observed. Based on this experience it was evident that the performance of the polyester coating on the canal surfaces was unsatisfactory and all further use of this material for this type of application was discontinued.

During these difficulties, other coating systems were being investigated in order to test repair selected failed areas in the canal. A phenolic type coating was chosen as the test repair material for blistered areas in the deep pit, based on the good performance of this coating in nuclear facilities at Idaho Falls. In addition, three other materials which included epoxy and phenolic epoxy as the base materials were chosen for test patches. A total of four coatings were therefore applied as test patches in December 1960. The epoxy and phenolic epoxy test patches encompassed an area of 25 square feet each. The test repair of the deep pit with the phenolic material was over 1000 square feet. Since surface preparation as well as application is extremely important, each of these coatings were applied under the direct supervision of the manufacturers' representatives. In this manner, correct application in accordance with the manufacturers' recommendations was assured.

The deep pit, which contained the test patches and test repairs, was filled with demineralized water and the area remained submerged for approximately a month. At the end of this time the pit was drained and the patch areas were inspected. The results of this inspection were as follows:

1. The phenolic-epoxy which was originally creamy in color had discolored to yellow. There were a number of pinholes and the coating lacked continuity. It was rough with sharp protrusions.

2. One of the epoxy base applications appeared to lack sufficient filler in the base coat and there were many ridges where the voids in the concrete were not built up to give continuity. However, the coating surface was smooth and blisters did not form during its submersion in demineralized water.

3. The other epoxy coating also lacked sufficient filler in the base coat; however, there was more continuity than that obtained for the coating in 2.

4. The phenolic coating contained numerous blisters which exuded a brown liquid when broken.

Since the test repair with the phenolic coating encompassed a much greater area than the test patches, an immediate investigation was undertaken. The manufacturer stated that an alcohol wash of the primer coat to remove excess catalyst was the main cause of difficulty. They had recommended the alcohol wash based on laboratory studies, which pointed out a reduction in time and coat compared with a detergent and water wash. However, during this field application the alcohol was not effective. A similar occurence had taken place at another installation, where as a result of the alcohol wash, delamination occurred with the top three coats separating from the primer in the blistered areas.
To determine the effect of various treatments on the test repair of the canal with phenolic, some of the blisters were removed and others were left "as is". In addition, the coating was removed entirely

in one area and the phenolic coating was applied using a water detergent wash. In conjunction with these tests, the manufacturer is conducting laboratory tests to determine corrective action.

Subsequent to the test patch inspection in the deep pit, the reactor pit was drained. Numerous blisters in the existing polyester coating were evident. Because of the poor condition of the polyester coating in the shroud pit and the numerous blisters in the reactor pit, a test repair was performed with an epoxy coating. The reactor pit will be inspected at an appropriate time. The results of this inspection in conjunction with manufacturers' tests and a review of coating applications at other installations will serve as a basis for corrective action.

<u>Summary</u>

Since the original application on the concrete surfaces, there has been a complete removal and reapplication of the original coating, test repairs, and test patches with various other coatings. Major problems with the first coating were lack of bonding to the concrete, blisters, and pinholes. Test repairs in some cases have been unsuccessful because of incomplete removal of catalyst from the primer coat. At the end of Seed 2 operations no specific coating has been selected and recommendations are being deferred until a complete examination of the test repairs and test patches can be performed after the Seed 2 - Seed 3 refueling period.

PART V

OPERATIONAL CHEMISTRY AND RADIOACTIVE CONTAMINATION EXPERIENCE

CHAPTER 1

OPERATIONAL CHEMISTRY-REACTOR PLANT

Introduction

This report presents the chemistry data and information, and discusses the highlights of activities in both operational chemistry and special test support in which the chemistry group played a significant part.

The highlights of reactor plant chemistry during Core 1 Seed 2 operations were:

(a) The change from natural lithium 92.5 a/o Li-7 and 7.5 a/o Li-6) to lithium-7 chemicals used for pH control. The lithium-7 chemicals and resin were 99.99 a/o Li-7. This reduced the amount of target atom present for the Li-6 (n,α) H-3 reaction and, hence, reduced the amount of tritium being processed to the environment;

(b) A special test which proved that the plant could be operated for extended periods without by-pass purification;

(c) The frequency of many of the routine control analyses were reduced because of the ease of control of certain chemistry conditions and the reliance which could be placed on certain measurements for routine fission product monitoring.

(d) The coolant sample trains which had been installed for the purpose of continuous monitoring of various reactor coolant water conditions were modified considerably due to excessive maintenance and the lack of necessity for continuous surveillance of the parameters.

One problem in coolant water chemistry occurred when the system was refilled at the conclusion of Seed 1 - Seed 2 refueling. The quantity of gases trapped and introduced into the system was large enough that the control rod drive mechanisms did not receive the necessary water lubrication. A seal weld was cut and the system was vented through the H-12 motor guide tube until the volume of trapped air was reduced to acceptable limits.

Reactor Coolant Sampling System

The reactor coolant is sampled at the inlet and outlet of the main purification system demineralizers. Samples are taken at full system pressure and approximately 120°F, via sample trains. Specific design information is given in Table V-A.

TABLE V-A

COOLANT SAMPLING SYSTEM DESIGN INFORMATION

Sample Point	Flow (gpm)	Length of Line (ft)	Fluid Velocity (ft/sec)	Reynolds Number	Delay Time (sec)
AC Loop	0.75 max	110	4.82	14000	58.6
	0.08	110	0.515	1500	250
BD Loop	0.75 max	220	4.82	14000	81
	0.08	220	0.515	1500	464

During operating periods, the sample flow is constant at 0.08 gpm. The maximum flow which can be achieved gives an 8-10 half-life decay for 7.5 second nitrogen-16 which is the major contributor to high energy gamma radiation during operation.

The sample trains shown in Figure V-1 were designed to monitor and record continuously certain coolant chemical and physical properties, as well as gross radioactivity, and to provide a point where samples can be removed from the coolant for laboratory investigation. The function of each component will be described briefly along with the operating experience gained to date.

The sample vessels are stainless steel vessels which can be inserted into the sampling system, isolated, and then removed to the laboratory. These vessels range in size from 10 to 500 milliliters. Experience during the operating period showed that sampling into a closed vessel is not necessary except when samples had to be taken at loop pressure for gas analysis. For all other analyses, samples are taken from an open valve directly into polyethylene bottles. During Seed 2 operations, the radiation level of a one liter sample in a polyethylene bottle was about 30 mr/hr gamma on contact with the container at sampling time.

The crud probe is designed to remove waterborne particulate matter at 100°F and loop pressure by filtration through a HA millipore filter. Material collected on the crud probe is weighed to determine crud concentration in the primary coolant; periodically crud probe samples are analyzed to determine the elemental and radiochemical characteristics of the crud. Because of the low levels of circulating crud in the reactor coolant (2-5 ppb), it was necessary to leave this probe in service for a period of one week to obtain a weighable quantity of crud. The radiation field produced by this integrated crud sample at the time the sample was removed from the sample train was in the general range of 0.8 to 14 R/hr gamma one inch from the sample.

The filter, which is separate from the crud probe, is used only when the probe is isolated and the sample train is on stream. All water passing through the sample train is filtered as a precautionary measure against fouling valves and instruments by particulate matter in the water. The filtering medium in the filter consists of a disposable teflon basket which is discarded to radioactive waste.

The pilot demineralizer is used to determine the average concentration for intermediate and long-lived radioisotopes in primary coolant solution. Flow from the crud probe passes through the demineralizer, which is charged with 150 cc of HOH form resin. At periodic intervals the resin is digested in an H_2SO_4-HNO_3 acid mixture with an activity determination made on the resulting solution. Activity levels of the resin are equated to an integrated average value for activity in solution by correcting for decay and rate of collection on the resin.

Effluent from the pilot demineralizer is also used to provide low conductivity water to the inlet of the thallium oxygen analyzer when an oxygen analysis is required. The thallium column measures oxygen concentration in water by the increase in conductivity of the effluent, the increase being directly proportional to oxygen concentration in the water.

Instrumentation in the Sample Train

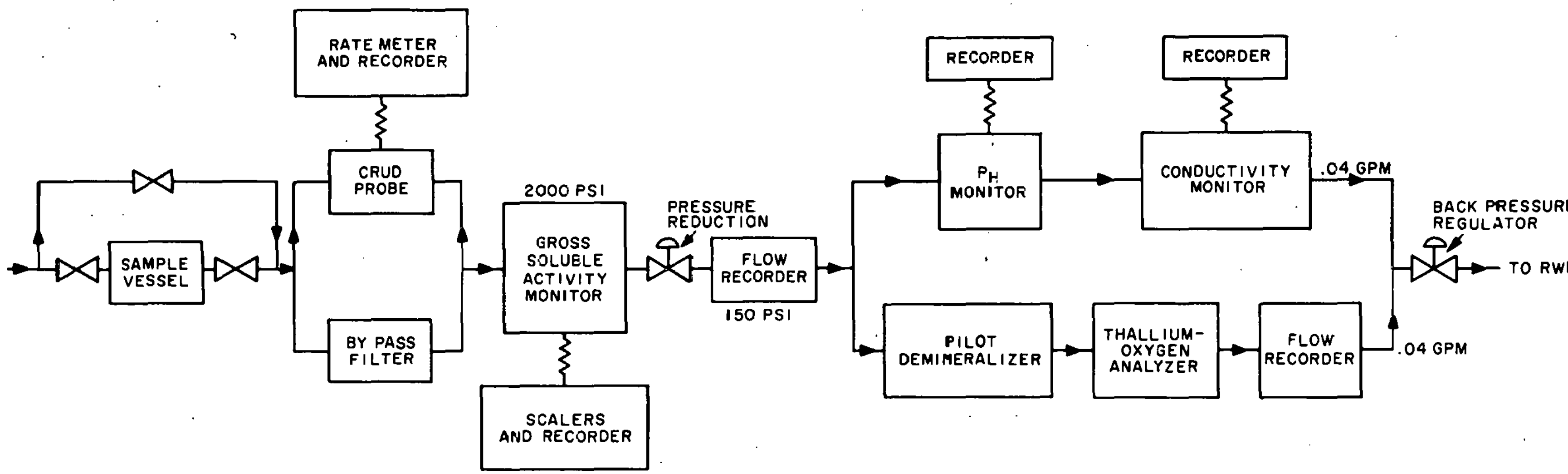
Development effort on primary coolant in stream monitoring devices was suspended during Seed 2 operations for a combination of one or more of the following reasons: (1) the instrument in question was adjudged to be operating successfully, (2) additional effort was not warranted since continuous monitoring was not required for plant protection, or (3) the instrument had a general unreliable operating history coupled with high maintenance requirements. The present status of the individual instruments is discussed below.

Crud Probe Monitor

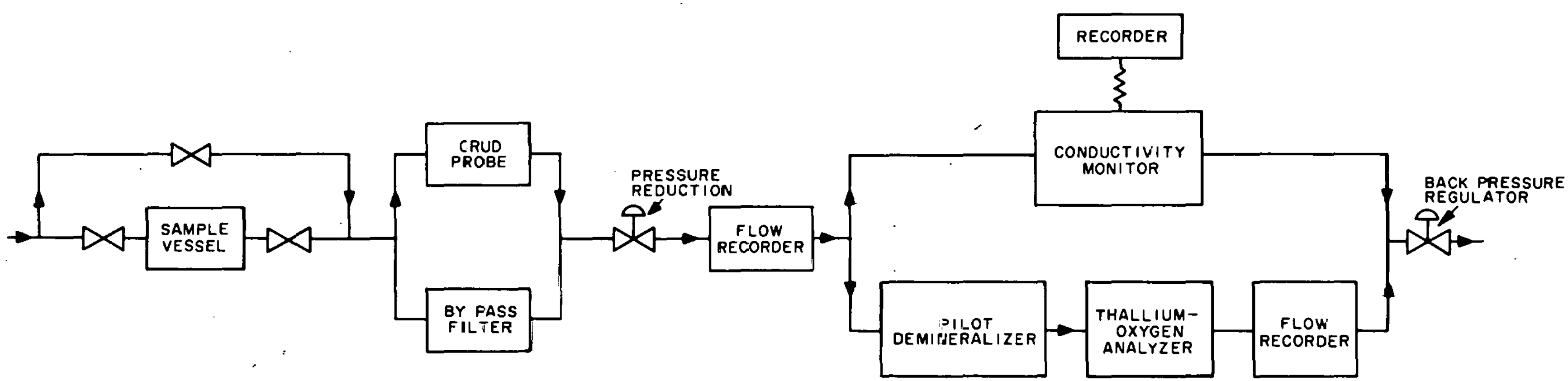
The crud probe monitor measures activity collected on a filter designed to remove water-borne particulate matter at approximately 100°F and full system pressure. Flow rate through the filter is 0.08 gpm. Because of problems with the ion chamber detector, the crud probe monitor operated for only a limited time during Seed 1 and Seed 2 operations. Of six detectors installed in the crud activity monitor, none functioned properly. Apparently, the detector is very sensitive to mechanical shock. Additional development effort on this instrument could not be justified since operational data indicate continuous monitoring of water-borne crud is not required for plant protection. The monitor and its associated rate meter and recorder have been removed from the sample train.

Gross Soluble Activity Monitor

The gross soluble activity monitor was designed to measure soluble activity in the primary coolant at temperatures in the range of 100°F and system pressure of 150 psi. This instrument had extremely high maintenance requirements and as a result only operated for a very short period during Seed 2 operations. There was an additional problem of activity build-up on the soluble activity cell which increased the background to levels which were at times 30 percent of the total counting rate. Since a sudden major increase in fission product activity from a large defect in a blanket fuel element would be detected by the FEDAL system, the need for operation of the soluble activity monitor could not be justified. For these reasons development efforts on the instrument were suspended, and the gross soluble activity monitor and its associated recorder have been isolated from the sample train.



SEED 1 SAMPLE TRAIN



SEED 2 SAMPLE TRAIN

Figure V-1. Block Diagram of Sample Trains.

Thallium Oxygen Analyzer

The thallium oxygen analyzer provides a means for monitoring oxygen concentration in the primary coolant down to the level of 5-10 ppb. This instrument performed reliably during Seed 2 operations. However, oxygen determinations on the primary coolant have been deleted during power operations, since excess hydrogen is available to combine with any oxygen which may be present. The detector was left in the sample train for stand-by use.

Primary Coolant Conductivity Monitor

Except for minor difficulties, the conductivity monitor performed satisfactorily and, in general, continuously during Seed 2 operations. One of the difficulties concerned itself with maintaining the conductivity cell in a fixed position. This cell was equipped with a retaining collar used to hold the cell in a fixed position by use of set screws. At design pressure, the position of the cell would shift, resulting in erroneous readings. The problem was resolved by welding the cell tubes to the retaining collar. A minor difficulty of the system was that the cells used in the equipment were not standard items requiring a long lead time for replacement.

pH Primary Coolant Monitor

This instrument never performed properly, even as adjusted by the factory representative. In test runs at the design pressure of 150 psi the cells would function from one to four hours. At reduced pressures of about 10 psig the instrument performed reliably for a maximum of 11 hours. The reason for failure was not established. Since the change in pH values of the primary coolant is a gradual as well as predictable process, continuous monitoring was considered unnecessary and developmental efforts were suspended. The pH meter and its recorder have been isolated from the system.

Reactor Coolant Water Conditions

The Shippingport PWR coolant is a high purity water system which is operated at high pH (9.5 - 10.5). The pH specification is maintained by periodic additions of lithium hydroxide monohydrate (99.99 a/o lithium-7) and by by-pass purification demineralizers which have been regenerated in the lithium hydroxide form. A summary of the reference water specifications is as follows:

	Conductivity (mmhos)	pH at 25°C	Lithium (ppm)	Dis. Oxygen* (ppm)	Hydrogen* (cc/kg S. T. P.)
Min.	7.0	9.5	0.3	0.14	15
Max.	75.0	10.5	2.3		60

* When the Reactor Coolant System is > 200°F

Maintenance of these specifications was accomplished without difficulty during Seed 2 operations. The ease of maintenance of water specification and the rather predictable trends in these conditions made possible a considerable reduction in the frequency of routine analytical chemistry laboratory support. The need for much of the in-stream monitoring equipment in the sample train was also precluded because of the nature of the changes experienced (see discussion of sample trains above).

At the conclusion of Seed 2 operation the frequency of many analyses had been reduced 50 percent or greater. Table V-B list for comparison the frequency of Seed 1 and Seed 2-Seed 3 laboratory analysis. The information in Table V-B lists the analytical first schedule requirements considered for the Seed 1 operation. Some reductions in frequency of various analyses occurred in Seed 1 and early in Seed 2 operation, particularly in the requirements for fission product analysis. However, only the initial and present schedules are presented for comparison.

Alkalinity

The reference water conditions for the reactor coolant were specified to be maintained at the same high pH as during Seed 1, that is, between 9.5 and 10.5. The pH was maintained by the addition of lithium hydroxide-monohydrate and by the lithium hydroxide resin in the purification system demineralizers.

During Seed 2 operations, a total of 2332 grams of lithium hydroxide monohydrate was added to the coolant in eleven increments over the 16 month period. Four additions totaling 733 grams were required during the no-purification period from June 23 to September 12, 1960. The fact that the pH was maintained by few chemical additions is indicative of a very high purity make-up water.

The relationship between pH, conductivity, and lithium was in good agreement throughout Seed 2. Conductivity proved to be a good check of the pH of coolant water conditions because of the high purity of the coolant. The maintenance of the pH reference water condition has been easily accomplished. The decrease in pH was very gradual and required only a weekly analysis for the purpose of pH control.

TABLE V-B. REACTOR PLANT SAMPLING AND ANALYSIS SCHEDULE CHANGES

Sample	Location	Determination	Frequency Seed 1	Frequency Seed 2-Seed 3	Comments
Coolant Charging	Primary Storage Tank	Conductivity	1/wk	1/mo	
		Dissolved O_2	1/wk	1/mo	
		pH	1/wk	1/mo	
		Cl	1/wk	1/mo	
Reactor Coolant	Laboratory analysis of 1AC or 1BD demineralizer influent	Conductivity	1/wk	1/wk	
		pH	1/wk	1/wk	
		Dissolved O_2	2/wk	--	Not specified when hydrogen present
		H_2	1/wk	Daily	
		Total gas	2/wk	Daily	
		Inert gas	1/wk	Daily	
		Lithium and NH_3	1/wk	1/mo	

TABLE V-B (Cont)

Sample	Location	Determination	Frequency Seed 1	Frequency Seed 2-Seed 3	Comments
	Demineralizer influent crud	Crud weight	Daily	1/wk	
		% Fe in crud	Daily	--	
		Gamma spectrum	--	1/wk	
		Specific activity	Daily	1/wk	
	Demineralizer effluent crud	Crud weight	1/wk	1/wk	
		% Fe in crud	1/wk	--	
		Gamma spectrum	--	1/wk	
		Specific activity	Daily	1/wk	
	1AC or 1BD demineralizer influent	Total gas activity	2/wk	2/yr	
		A^{41}	2/wk	2/yr	
		H_3	1/wk	Every 1000 EFPH	
Reactor Coolant	Laboratory analysis of 1AC or 1BD demineralizer influent	Gross γ activity (15 min)	Daily	Daily	
		120 hr count	Daily	Daily	
		Cs^{138}	Daily	3/wk	
		Kr^{88}	Daily	--	
		$I^{131,133}$	3/wk	1000 EFPH	
		$Cs^{136,137}$	Daily	1000 EFPH	
		$Br^{83,84}$	3/wk	--	
		Xe^{133}	1/wk	1000 EFPH	
		Gross I decay curve for I^{131}, $I^{132,133}$	3/wk	--	
		Decontamination factor	Daily	1/wk	
	Pilot demineralizer and crud probe	85% gamma ray balance, major activity, $Co^{60,58}$ Fe^{59}, Cr^{51}, Mn^{54}, HF^{181}, Zr^{95}	*	1000 EFPH	*Seed 1 - analyzed when activity had moved 1/3 of column length.

TABLE V-B (Cont)

Sample	Location	Determination	Frequency Seed 1	Frequency Seed 2-Seed 3	Comments
Canal Water	Demineralizer inlets and outlets	Conductivity	2/wk	1/wk	
		pH	2/wk	1/wk	
		Turbidity	--	1/wk	
		Gross γ activity	Daily	1/wk	
Component Cooling Water	Component cooling water pump suction	pH	2/mo	1/wk	
		CrO_4	2/mo	1/wk	
		Conductivity	--	1/wk	
		Cl	--	1/wk	
		Gross γ activity	2/mo	1/wk	
Neutron Shield Tank	Eductor	pH	*	2/yr	*As frequently as access to sample point is possible.
		CrO_4		2/yr	
		Gross γ		2/yr	
		K^{42}		--	
Hair Pin Loop	Purification cubicle	Activity level survey I^{131}, Co^{58}, Cs^{136} Cs^{137}, Te^{132}, Ba^{140}, $Sr^{89,90}$, Ce^{144}, Fe^{59}, Co^{60}, Zr^{95}, Ta^{182}, Ta^{183}, U	See comment		Performed as required for the AEC test program.
Valve Operating System	Drain on water flush	Dissolved O_2	1/wk	--	Dropped from schedule.

Dissolved Hydrogen and Total Gas

The reference water specification for dissolved hydrogen in the reactor coolant is 15-60 cc/kg of water at STP. Hydrogen is used to control the water decomposition reaction and to combine with any oxygen charged to the system. The hydrogen specification was maintained without difficulty. Special testing was performed to determine the rate of hydrogen loss from the reactor coolant during steady-state operations with water seals in the pressurizer steam space relief valves and evaluate the feasibility of adding hydrogen at the inlet of the purification system ion exchangers as added protection against introducing undissolved hydrogen into the main coolant pumps.

The time required for the hydrogen concentration to decrease to a value of one-half the original concentration is shown below. These values were determined by calculating the slopes, from a typical curve shown in Figure V-2. The initial concentration is the concentration immediately after addition; however, it is an extrapolated value taken from each curve coincident with the time of addition.

Injection Point	Time
Inlet - 1AC ion exchanger	153 hours
Inlet - 1BD ion exchanger	172 hours
Outlet - 1BD ion exchanger	118 hours
Outlet - 1AC ion exchanger	122 hours

The difference in hydrogen loss rates in the above table can be attributed to the number of main coolant loops in service during each period. Hydrogen loss from the reactor coolant to the pressurizer steam space increases with increasing pressurizer spray flow. The pressurizer spray flow rate increases with increasing reactor vessel pressure drop which depends on the number of loops in operation. The station was on three-loop operation during the test runs when hydrogen was added to the inlet of the demineralizers, and on four loop operation during the test runs when hydrogen was added to the outlet of the demineralizers. Therefore, if a correction is added for the different pressurizer flow rates during each test run, the rate of hydrogen loss for each of the four test runs is approximately the same.

The decontamination factors of both the 1AC and 1BD purification system ion exchangers were not affected by the addition of hydrogen to the inlet side of the ion exchangers.

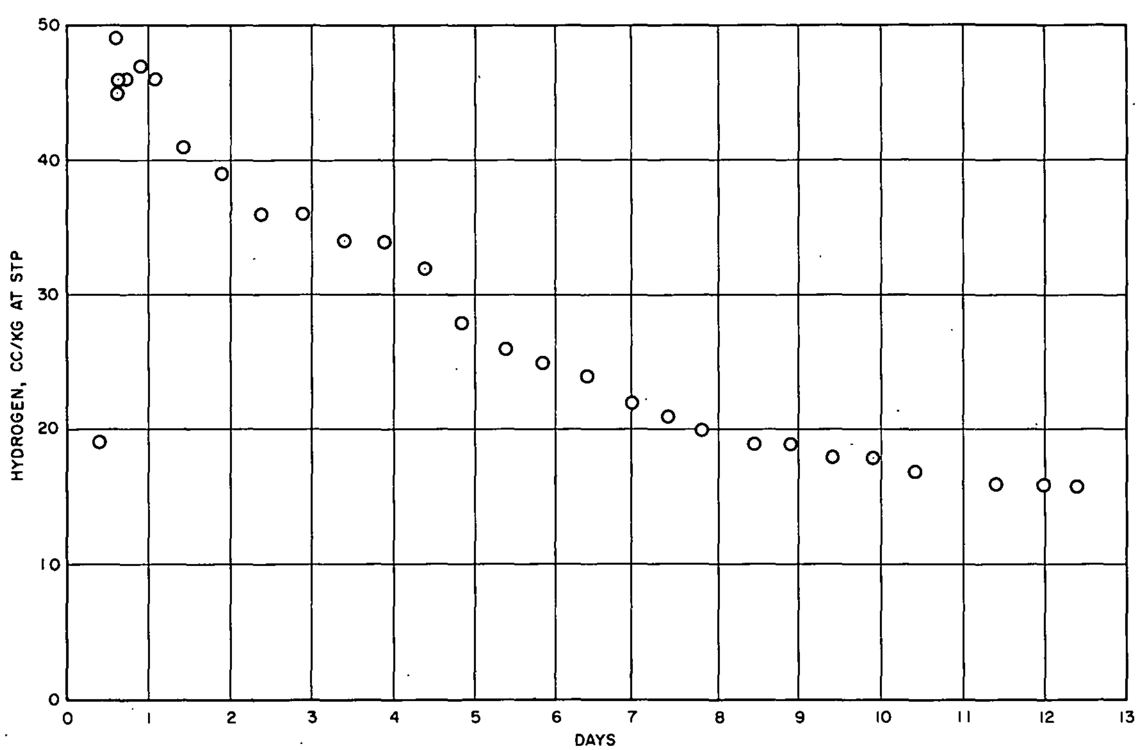


Figure V-2. Hydrogen Distribution Test (December 13, 1960 to December 25, 1960).

The total concentration of inert gases dissolved in the coolant ranged from 5 to 10 cc/kg of water during Seed 2 operation with no incidents involving total gas content during normal operations.

At the conclusion of Seed 1 refueling and prior to the beginning of Seed 2 operations, gas analyses and pressurization of the reactor plant indicated that approximately 150 cu ft of undissolved gas was present in the reactor vessel head. This situation probably resulted from air being trapped in various sections of the reactor plant during the draining and filling of the loops and pressure vessel. Incomplete venting resulted in the high quantity of undissolved gas. The mechanism requirements for undissolved gas in the reactor head area is that the maximum quantity should be less than33 cu ft.

The H-12 motor guide tube seal weld was cut and the system was bled until the total dissolved gas was reduced to 33 cc/kg. The system pressure was maintained at 400-500 psig to keep the contained gases in solution. The dissolved gas concentration was further reduced by operation of the pressurizer spray and venting through the pressurizer steam relief system. A procedure has been written to prevent reoccurence of the above problem.

Dissolved Oxygen

During all periods of normal operation, the <0.14 ppm dissolved oxygen specification was maintained with the dissolved oxygen ranging from <0.005 ppm to 0.010 ppm. The analytical requirements for dissolved oxygen during periods when there was an excess of hydrogen were considered unnecessary and were removed from the analytical schedule (see Table V-B).

These changes were made to avoid the difficulty which had arisen during the increase in main steam ammonia discussed above. The change in oxygen specification was made to avoid the unnecessary discarding of high purity water containing small amounts of oxygen. The oxygen in the charging water would be consumed since the coolant would contain excess hydrogen.

Reactor Plant Storage Tank

Reference water specifications were maintained in the coolant make-up water almost without exception, during the Seed 2 operating period. One incident occurred during February, 1961. The main and auxiliary system ammonia concentrations increased because of an increase in combined ammonia (albuminoid ammonia) in the Ohio River water. The auxiliary steam system supplies steam service for the steam blanket on the primary water storage tank. Frequent conductivity alarms were received as the conductivity approached the 1.5 mmhos alarm point. The conductivity of primary charging water, when corrected for the conductivity of the dissolved ammonia, gave very low values indicating that the increase in conductivity was entirely accounted for by the increase in the ammonia content of this water.

To reduce the ammonia content of the main steam system and hence the conductivity of the charging water, the condensate demineralizer (polishing demineralizer) was placed in service in the condensate system, and the ammonia of the main and auxiliary water was less than one mmhos within 24 hours.

Near the end of Seed 2 life, the conductivity specification of the primary water storage tank was changed from 1.5 to 2.5 mmhos and the dissolved oxygen specification suspended. Because of ease of maintenance of the dissolved oxygen specification, the lower dissolved oxygen specification has been maintained although not required. This change in conductivity and dissolved oxygen specifications was justified on the basis of successful operation of other plants with make-up water of such quality.

Component Cooling Water and Neutron Shield Tank

The reference water specifications of the need for the component cooling water and neutron shield tank were maintained without difficulty during Seed 2 operation. The activity in the neutron shield tank after 5 days decay is entirely accounted for by Cr^{51} and a small acount of Co^{60} activity. The Cr^{51} activity ranged from 11,000 dpm/ml ($4.9 \times 10^{-2}\ \mu c/ml$) to 40,000 ($1.8 \times 10^{-2}\ \mu c/ml$) dpm/ml and was due to the Cr^{50} (n, γ) Cr^{51} reaction, the target atom being supplied in the chemical, K_2CrO_4. The source of cobalt for the Co^{59} (n, γ) Co^{60} reaction was determined to be a minor contaminant of the treatment chemical. The activity of Co^{60} was about 40 dpm/ml ($1.8 \times 10^{-5}\ \mu c/ml$).

Coolant Purification System

The performance of the coolant purification system was checked daily during operating periods by measuring the gross nonvolatile gamma activity 15 minutes after sampling at the inlet and outlet of the purification demineralizers. The initial charge of lithium hydroxide resin remained in service from December 1957 until November 6, 1960 with the resin decontamination factor never decreasing <25 for four consecutive days, the criterion for replacement.

In November 1960, the resin was discharged and replaced with Li^7 form resin to reduce the concentration of tritium in the primary coolant. Primary coolant pH at PWR is controlled in the range of 9.5 to 10.5 by use of lithium hydroxide. Originally, natural lithium which contains 7.5 percent Li^6 and 92.5 percent Li^7 was used for this purpose. This resulted in a high tritium concentration in the primary coolant from the Li^6 (n, α) H^3 reaction. Tritium decays with a half-life of 12.26 years by emitting 18 Kev beta rays. While tritium has a short biological half-life (19 days), its long radiological half-life, and the fact that tritium is not removed by the radioactive waste disposal system, makes it desirable to reduce the tritium discharged from PWR by limiting tritium production. To accomplish this, the natural lithium resin at PWR was replaced with Li^7 OH resin. The lithium in the replacement resin contained 0.0068 ± 0.0005 a/o concentration of Li^6. Any subsequent additions of Li^7 OH to the coolant employed Li^7 OH containing less than 0.01 a/o Li^6 in total lithium. Lithium-7 hydroxide monohydrate is commercially available from the U.S. Atomic Energy Commission.

Following the changeover to Li^7 resin, tritium concentration in the primary coolant was reduced by approximately a factor of 100, as shown in Table V-C. The low tritium concentration which was finally achieved indicates that there was little holdup of Li^6 in the primary system when the changeover was made to Li^7 resin.

TABLE V-C. TRITIUM CONCENTRATION IN THE PRIMARY COOLANT AT PWR

Date	Coolant Sample Tritium ($\mu c/l$)	Date	Coolant Sample Tritium ($\mu c/l$)
6-6-60	221	12-30-60	16.1
9-27-60	281	1-6-61	18.8
10-9-60	241	1-13-61	6.7
11-7-60	128	1-24-61	5.9
11-21-60	86	2-6-61	13.2
12-8-60	23	2-12-61	3.2
12-11-60	19.5	2-28-61	1.5
12-18-60	18.7	4-8-61	1.3
12-23-60	18.5	4-26-61	2.3

Base Radiation Levels During Seed 2 Start-up

During initial Seed 2 life a radionuclide base level determination was made to determine if the new seed contained defective or excessively contaminated elements, and to provide a fission product base level for comparison in evaluating core integrity during subsequent power operations. Analyses were performed at 65.2 EFPH and 89.9 EFPH after a steady-state run at 100 percent power for approximately 48 hours. The major activities reported were N^{13}, F^{18}, Na^{24}, W^{187}, Mn^{54}, Mn^{56}, Cu^{64}, Cs^{138}, and radioactive iodine. A total of 95 percent of the gross gamma activity at one hour after sampling was accounted for in the activity balance. Results of the analyses are presented in Tablc V-D.

The largest short-lived contributor to the gross gamma activity was Cu^{64}. The high copper activity is probably the result of a flow integrator failure which occurred during Seed 1 operation, resulting in the release of copper to the primary system. There is evidence that the concentration of copper in the system is decreasing with time in operation based on elemental analysis of composites of Seed 1 and Seed 2 crud. The Seed 1 crud sample had a copper concentration of 2.3 percent versus 1.7 percent for the Seed 2 crud sample.

Other major contributors to short-lived activity included N^{13} and F^{18}. Because of their short half-lives, the short-lived activities are not a problem with respect to plant maintenance.

TABLE V-D. SPECIFIC ACTIVITY FOR CORE I SEED 2 OPERATION AT PWR

A. Short-Lived Nuclides

Nuclide	Half Life	Sample Time (EFPH)	dpm/ml x 10^{-3}	μc/ml x 10^{4}	Sample Description
N^{13}	10.1 m	65.2	66.7	300	Degassed coolant, demineralizer inlet
F^{18}	1.87h	65.2	28.0	126	Degassed coolant, demineralizer inlet
A^{41}	1.82h	89.9	8.5	38.7	Unfiltered coolant, demineralizer outlet
Na^{24}	15.0h	65.2	4.2	19.1	Degassed coolant, demineralizer inlet
Mn^{56}	2.59h	65.2	0.705	3.21	Degassed coolant, demineralizer inlet
I^{133}	21.0h	89.9	2.54	11.6	Unfiltered coolant, demineralizer inlet
I^{135}	6.7h	89.9	8.35	38.0	Unfiltered coolant, demineralizer inlet
Kr^{87}	3.2m	89.9	1.44	6.65	Unfiltered coolant, demineralizer outlet
Kr^{88}	2.8h	89.9	9.2	41.9	Unfiltered coolant, demineralizer outlet
Sr^{91}	9.7h	89.9	0.002	0.009	Filtered coolant, demineralizer inlet
Sr^{92}	2.7h	89.9	0.009	0.040	Filtered coolant, demineralizer inlet
Ba^{139}	8.5m	89.9	0.297	1.35	Filtered coolant, demineralizer inlet
Ba^{139}	8.5m	89.9	0.028	0.127	Primary crud, demineralizer inlet
Cs^{138}	32.0m	65.2	11.3	51.4	Degassed coolant, demineralizer inlet
W^{187}	24.0h	65.2	2.55	11.6	Degassed coolant, demineralizer inlet
Cu^{64}	12.8h	65.2	57.0	257.0	Degassed coolant, demineralizer inlet

NOTE: All values corrected to sample time.

B. Long-Lived Nuclides

Nuclide	Half Life	Sample Time (EFPH)	dpm/ml.	μc/ml x 10^{7}	Sample Description
Cs^{136}	13d	89.9	0.8	3.64	Filtered coolant, demineralizer inlet
Cs^{136}	13d	89.9	0.86 ± 0.035	3.91	Primary crud, demineralizer inlet
Cs^{137}	28.6y	89.9	9.2	41.9	Filtered coolant, demineralizer inlet
Cs^{137}	28.6y	89.9	0.23 ± 0.035	1.05	Primary crud, demineralizer inlet
I^{131}	8.05d	89.9	189	819	Unfiltered coolant, demineralizer inlet
Sr^{89}	54d	89.9	2.7 ± 0.06	12.3	Filtered coolant, demineralizer inlet
Sr^{89}	54d	89.9	0.46 ± 0.035	2.09	Primary crud, demineralizer inlet
Sr^{90}	28y	89.9	Not detectable in filtered sample or primary crud, demineralizer inlet		
Ba^{140}	12.8d	89.9	Not detectable in filtered coolant, demineralizer inlet		
Ba^{140}	12.8d	89.9	7.4	33.7	Primary crud, demineralizer inlet
Mn^{54}	300d	65.2	205	933	Degassed coolant, demineralizer inlet

NOTE: All values corrected to sample time.

Fission Products

The routine monitoring of fission products was continued throughout Seed 2 life. The determination of changes in core integrity is based on changes in fission product distribution and by peaking of long-lived fission products during power transients. The fission product data also serve as a guide in the evaluation of the failed element detection and location system data.

In an effort to determine the presence of defective fuel elements, iodine fission products were followed closely at start-ups. The high peak values of the specific activities of I^{131} and I^{133} in the reactor coolant system indicate that one or more defective blanket fuel elements were present in the core. The peak values of I^{131} were as high as fifteen times greater than normal concentrations at 100 percent power.

The quantity of long-lived fission products released is dependent on the previous irradiation history of the fuel and the length of the shutdown, i.e., half-life of the nuclide analyzed. Because of these factors, the iodine-133 (20.8 hour half-life) peak is not as definitive as iodine-131 (8.08 day half-life).

During power transients, Cs^{134} (2.3 year half-life) was observed. Normally, Cs^{134} is not detectable in the coolant. The appearance of Cs^{134} may be a further indication of failed blanket assemblies. The levels of Cs^{134} activity during these periods of power transients range from less than detectable to 165 dpm/ml (7.43×10^{5} μc/ml).

The activity of fission products during steady-state power operation did not change significantly during seed lifetime. The average value of fission products analyzed over each 500 EFPH operating period is presented in Tables V-E and V-F.

Activation Products

The activation products are introduced into the system principally from impurities in the makeup water and from corrosion and wear.

Argon activity in the primary coolant comes from atmosphere argon (air contains 0.9% A^{40}) dissolved in makeup water. Degassing operations remove most but not all of the dissolved gases in the makeup water. Through neutron capture, A^{40} forms A^{41} which is a beta-gamma emitter. An increase in A^{41} is noted following the addition of a very large quantity of makeup water. The highest level during Seed 2 was 1.45×10^{5} dpm/ml (6.53×10^{-2} μc/ml) and the lowest level 9.60×10^{2} dpm/ml (4.32×10^{-4} μc/ml) with an average value of about 2.80×10^{3} (1.24×10^{-3} μc/ml).

A weekly investigation of waterborne particulate matter was performed during Seed 2 operations. These analyses are performed to follow corrosion and wear characteristics of the system, i.e., crud concentration in the primary coolant is related to corrosion release rates while cobalt concentration in the crud comes primarily from wear on stellite surfaces. The principal contribution to long-lived radiation levels of the reactor system comes from the activation of Co^{59} to form Co^{60}.

Crud specific activity increased from about 1.00 x 10^7 cpm/mg at the beginning of Seed 2 to a level of about 3.0 x 10^7 cpm/mg at the end of Seed 2 life. Approximately half the increase resulted from the buildup of intermediate-lived nuclides, such as Co^{58}, Fe^{59}, Cr^{51}, Hf^{181} and Zr^{95}, with the remainder coming from the increase in Co^{60} specific activity. This increase was, in the main, the result of additional irradiation time on the system, since elemental crud analyses presented in Table V-G indicate that the elemental characteristics of the crud or relative concentration of target atoms remained unchanged from Seed 1 to Seed 2 operations. Crud sample analysis presented in Table V-E were obtained by X-ray fluorescense and optical spectrography for elemental Co, Fe, Cr, Ni, Cu, and Mn. Except for crud bursts encountered during power start-ups, crud concentrations remained low in the range of 2-5 ppb during Seed 2 operations. The maximum crud concentration of 45 ppb was observed during a power start-up at the beginning of the no purification run.

An upward trend was observed in Hf^{181} activity. The specific activity of Hf^{181} increased from 1.7 x 10^5 at the beginning of Seed 2 life to a peak value of 5.6 x 10^6 dpm/mg near the end of Seed 2. A comparison of Seed 1 and Seed 2 results is shown in Figure V-3 as a plot of activity versus time. The upward trend in Hf activity may indicate a change in corrosion rate of the hafnium or increased wear on the hafnium rubbing shoes. Excessive wear or corrosion of the Hf control rods would be cause for concern, since this would result in increased plant radiation levels. However, on the basis of analysis of pipe wall deposits from the hairpin loop, there was no significant increase in Hf released to the system, since Hf activity accounted for only 0.1 to 0.5 percent of the total radiation level outside the pipe. Representative Hf activity data from crud analysis are presented in Table V-H.

Hairpin Loop

During Seed 2 the test pipe section, referred to as the hairpin loop, was removed and analyzed to determine the buildup of activity on the plant surfaces. At the beginning of Seed 2 lifetime, the previous section was replaced with a new section. At this removal, September 12, 1960, a section operating for 1476.0 EFPH, and one with 5651.7 EFPH, was analyzed. The 1476.0 EFPH section had accumulated hours only during Seed 2 operation while the 5651.7 EFPH section had accumulated 3403.5 EFPH during Seed 1 plus 2248.2 EFPH during Seed 2. (Refer to Reference 1 for location of test pipe section in the system.)

The pipe surface was brushed with a nylon brush and washed simultaneously with a high-speed stream of water. The crud was weighed and its deposition calculated at 0.2 mg/cm^2 on the new section (1476.0 EFPH) and 1.64 mg/cm^2 on the old section (5651.7 EFPH). Radiation surveys on contact, in the center of the schedule 160 2-inch pipe, using a Jordon probe-type monitor, showed a maximum radiation field of 100 mr/hr on the new section and 175 mr/hr on the old section. The gross γ activity on the new section was 1.04 x 10^6 cpm/cm^2 as compared with the old section 2.10 x 10^6 cpm/cm^2. The results of the analysis indicated that the highest rate of activity buildup occurs during initial exposure of the loop piping to coolant flow, indicating that the buildup mechanism may be related to corrosion rate. Activity buildup on the hairpin loop was below that observed on the main loop piping probably because of a distance effect.

TABLE V-E

SPECIFIC ACTIVITY X 10^{-3}, dpm/ml* CORRECTED TO 4 LOOP FULL PURIFICATION

EFPH	Cs^{138}	Cs^{139}	Kr^{85m}	Kr^{87}	Kr^{88}	Xe^{135}	I^{131}	I^{133}
500	9.0	--	1.1	1.1	9.2	4.2	0.1	1.2
1000	10.6	3.5	3.0	2.5	10.8	10.1	0.1	1.2
1500	11.4	--	3.9	3.3	12.7	15.1	0.4	1.3
2000	12.3	2.5	2.8	3.4	15.3	8.6	0.3	1.4
2500	12.7	0.9	2.3	1.9	14.1	8.9	0.2	1.3
3000	13.6	7.3	0.9	0.6	12.1	4.2	0.2	1.7
3500	10.2	6.4	0.9	--	16.0	3.6	0.1	1.3
4000	11.8	1.0	1.9	1.4	8.5	4.9	0.3	1.9
4500	10.0	4.7	1.3	1.2	7.5	6.2	0.2	1.5
5000	11.9	13.4	1.7	0.4	6.7	7.1	0.2	1.6
5500	10.2	10.0	1.0	0.5	7.2	4.0	0.1	2.0
6000	10.0	9.6	0.9	1.0	6.9	4.9	0.2	1.5
6500	10.6	6.4	1.3	1.4	5.8	5.1	0.2	2.3
7000	9.9	6.5	1.0	1.1	4.8	5.0	0.1	1.4
7500	9.6	8.0	0.9	1.3	5.4	5.6	0.1	1.6

*For conversion to the microcurie unit, multiply the tabular value by 4.5×10^{-4}
(dpm-ml x 10^3 x 4.5×10^{-4} = μc/ml)

TABLE V-F

REPRESENTATIVE FISSION PRODUCTS-CORE I SEED 2

Date	Reactor Power Level (%)	(Hrs.)	Specific Activity (dpm/ml x 10^{-3})** Cs^{138}	Kr^{88}	Kr^{85}	Kr^{87}	I^{131}	I^{133}	Xe^{133}	Xe^{135}	Sr^{91}	Sr^{92}	Br^{82}	Br^{83}
6-6-60	100	357	14.0		1.8	1.30			55.0	5.8				
6-10-60	100	453	13.0	11.0	2.4	1.80	0.25	2.3	61.0	7.6	0.22	0.10		
*6-30-60	100	119			4.97	4.32								
*7-6-60	100	332	15.0	14.0			14.1	13.0	49.3	16.5				
*8-8-60	100	13	13.2				38.8	21.3						
*8-9-60	100	37		19.1	4.78	3.92			70.5	16.9	0.09	0.18	1.35	3.35
*8-23-60	100	103			2.95	5.10	35.6	20.7	73.2	10.1			1.98	3.98
*9-6-60	100	438		19.3									2.04	4.21
10-4-60	100	200		13.3	0.89	0.86	0.50	5.86	13.2	3.00				0.84
10-24-60	100	261		17.1										
11-10-60	100	55					0.30	3.30						0.27
2-10-61	100	132	13.8	7.79	1.88	0.49	0.13	1.58	51.0	8.00				
3-7-61	100	350	9.94	7.20	1.08	1.09								0.33
3-14-61	100	530		7.45	1.15	1.15	0.14	1.38	37.0	5.10				0.52
4-28-61	100	48	13.4	6.90			0.24	2.47						
5-23-61	100	42		5.50			0.14	1.80						0.69
6-9-61	100	32		6.77	1.00	1.45			45.7	6.32				
7-11-61	75	27	8.78	7.90	1.60	3.47	0.19	1.47	20.7	9.34				0.33
8-4-61	50	129	5.33	2.52	1.07	1.21	0.15	0.99	10.8	6.20				

*No Purification

**For conversion to the microcurie unit multiply the tabular value by 4.5 x 10^{-4}
(dpm-ml x 10^{-3} x 4.5 x 10^{-4} = μc/ml)

TABLE V-G
X-RAY FLORESCENCE

Element	Seed 1	Seed 2
Co	0.3	0.3
Fe	51.6	56.0
Cr	0.5	1.2
Ni	5.9	4.6
Cu	2.3	1.7
Mn	0.7	0.8

Purification Shutoff Test

During Seed 2 operations the reactor system was operated from June 23, 1960 to September 12, 1960 for a total of 1478 EFPH with the purification system isolated to determine the effectiveness of purification flow in reducing the buildup of long-lived radiation levels in the reactor system. Evaluation of this test was based on data taken in conjunction with operational and test requirements; this consisted of (1) radiation levels of the primary loop piping, (2) activity buildup on the reactor vessel head, (3) crud levels in the primary coolant, (4) crud specific activity, (5) corrosion product activity in the primary coolant, (6) fission product levels in the primary coolant, (7) 15 minute gross non-volatile gamma activity, and (8) pressure drop across the core. These data are shown as functions

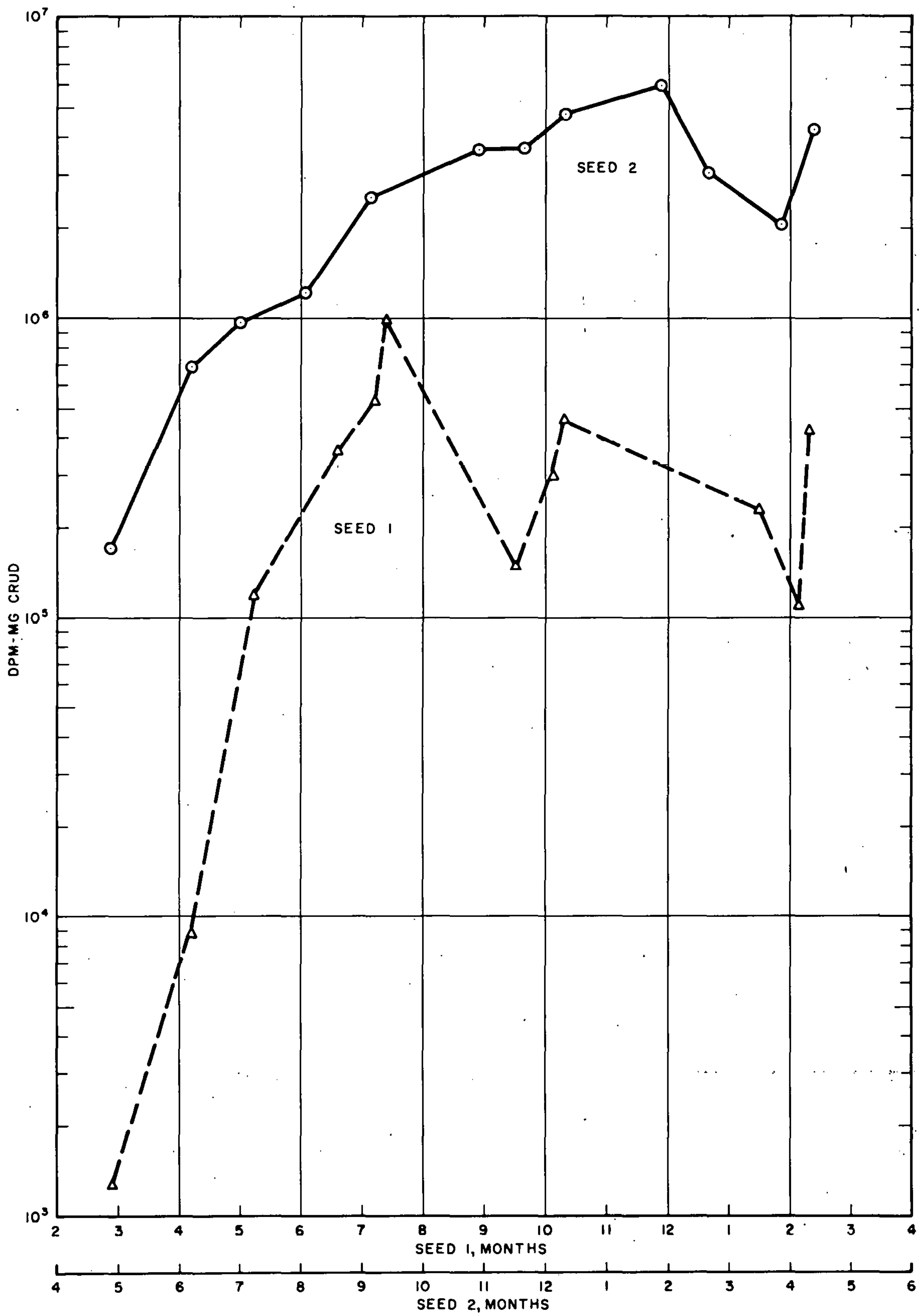


Figure V-3. Comparison of Seed 1 - Seed 2 Hafnium Activity.

of EFPH in Figures V-4 through V-8, except for data on corrosion product activity in the primary coolant which are presented in Tables V-I, V-J and radiation survey which is presented in Part V, Chapter 3. These tables include Seed 1 data in order to show that corrosion product activity levels in the primary coolant during the test period remained within the general scatter observed during operation with purification. The peak corrosion product activity level observed on June 24, 1960 occurred one day after the purification system had been isolated. The specific activity of Co^{60} at this time was 690 dpm/ml compared to 50-100 dpm/ml for normal operations. A sharp increase of this type in primary coolant activity levels is characteristic of crud bursts. Among other reasons, crud bursts occur as a result of temperature or system transients. Operating history of the reactor system in the immediate period preceding June 24, 1960 included a start-up to power and two safety insertions.

Within the scatter of data the plant as a whole showed no demonstrable change in the trend of long-lived activity buildup during operation without purification. Gross degassed gamma activity levels and specific activities of the soluble fission product species with a half-life appreciably longer than four hours increased when the purification system was isolated. This, however, did not present an operational problem. Upon return to purification flow, these activities were quickly reduced to pre-test levels. On the basis of these data it is considered that operation without purification has been proven feasible. However, the purification system is useful in reducing soluble activity and in providing the ability to recover from an inadvertent addition of poor quality charging water.

TABLE V-H

SPECIFIC ACTIVITY OF CRUD*

Source Date	Fe^{59}	Co^{58}	Co^{60}	Mn^{54}	Zr^{95}	Hf^{181}	Cr^{51}	% of Total γ Activity
BD-BIX 5/23/60	6.44×10^5		1.56×10^7	1.31×10^6	2.58×10^4	1.70×10^5	5.70×10^5	78.7
BD-BIX 6/25/60	2.60×10^6		1.58×10^7	1.81×10^6	1.10×10^5	6.80×10^5	2.85×10^6	79.8
BD-BIX 7/22/60	2.87×10^6	1.96×10^6	2.49×10^7	2.07×10^6	2.33×10^5	9.54×10^5	1.75×10^6	88.4
BD-BIX 8/29/60	2.64×10^6	2.54×10^6	1.85×10^7	2.01×10^6	9.08×10^4	1.22×10^6	8.10×10^5	73.2
AC-BIX 9/26/60	2.61×10^6	1.69×10^6	1.61×10^7	1.65×10^6	2.00×10^5	2.48×10^6	1.32×10^6	73.5
AC-AIX 10/10/60	2.58×10^4		1.73×10^6	7.54×10^5		6.75×10^4		70.5

*In dpm/mg. For conversion to the microcurie unit multiply the tabular value by 4.5×10^{-7} (dpm-mg $\times$ 4.5×10^{-7} = μc/mg)

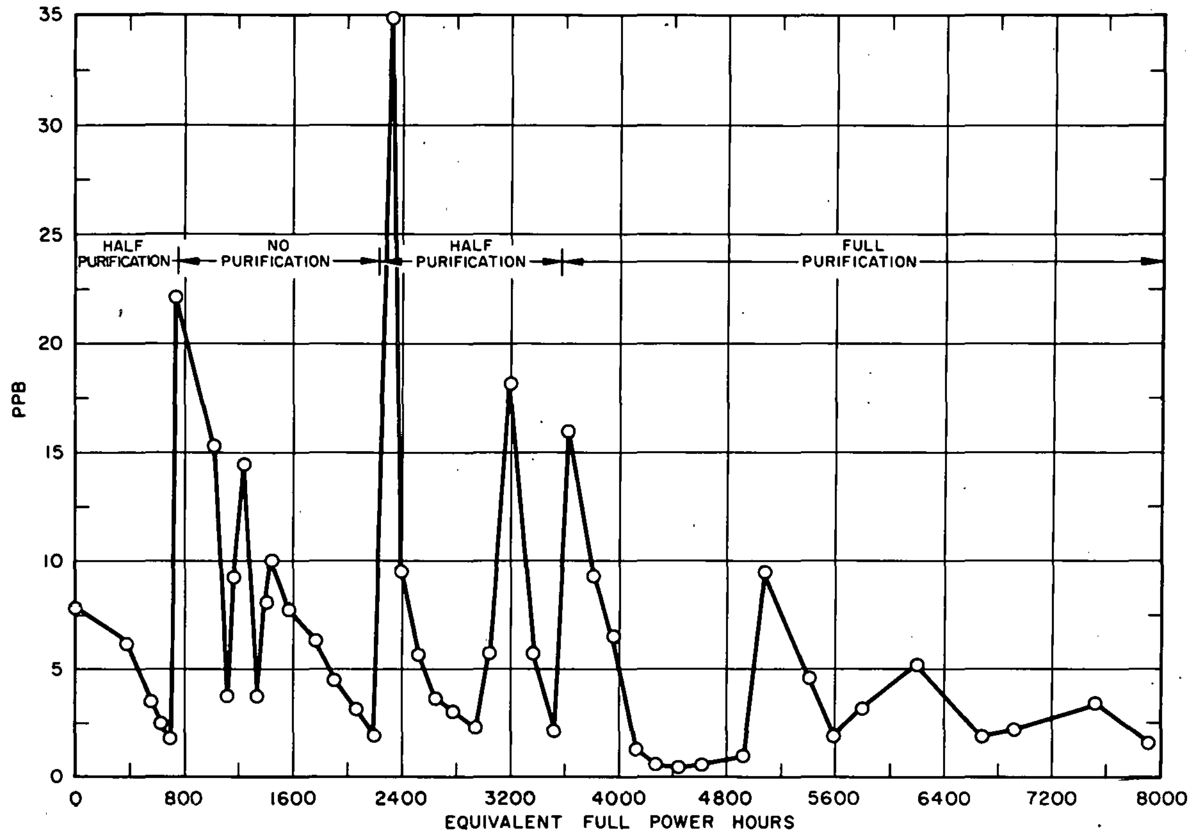


Figure V-4. PWR Crud Concentration in Primary Coolant (ppb).

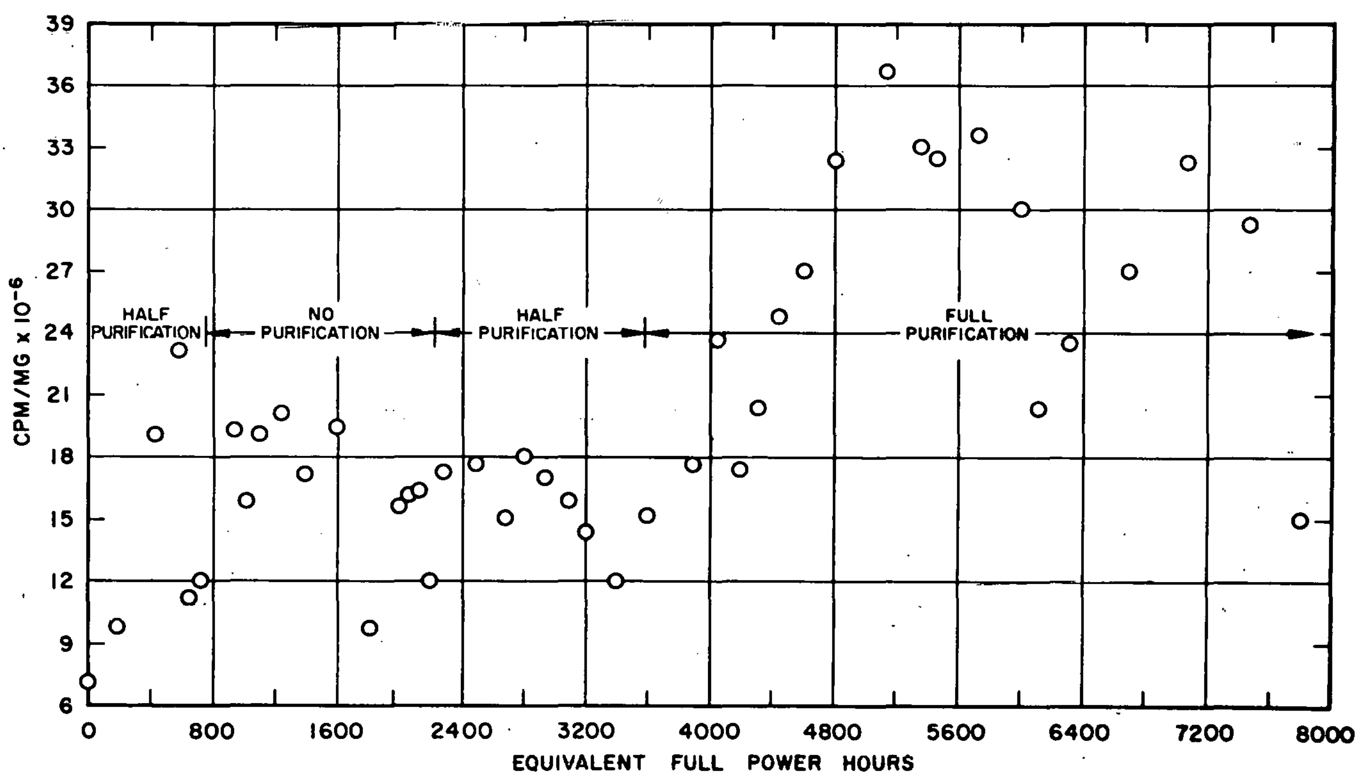


Figure V-5. PWR Crud Specific Activity (cpm/mg).

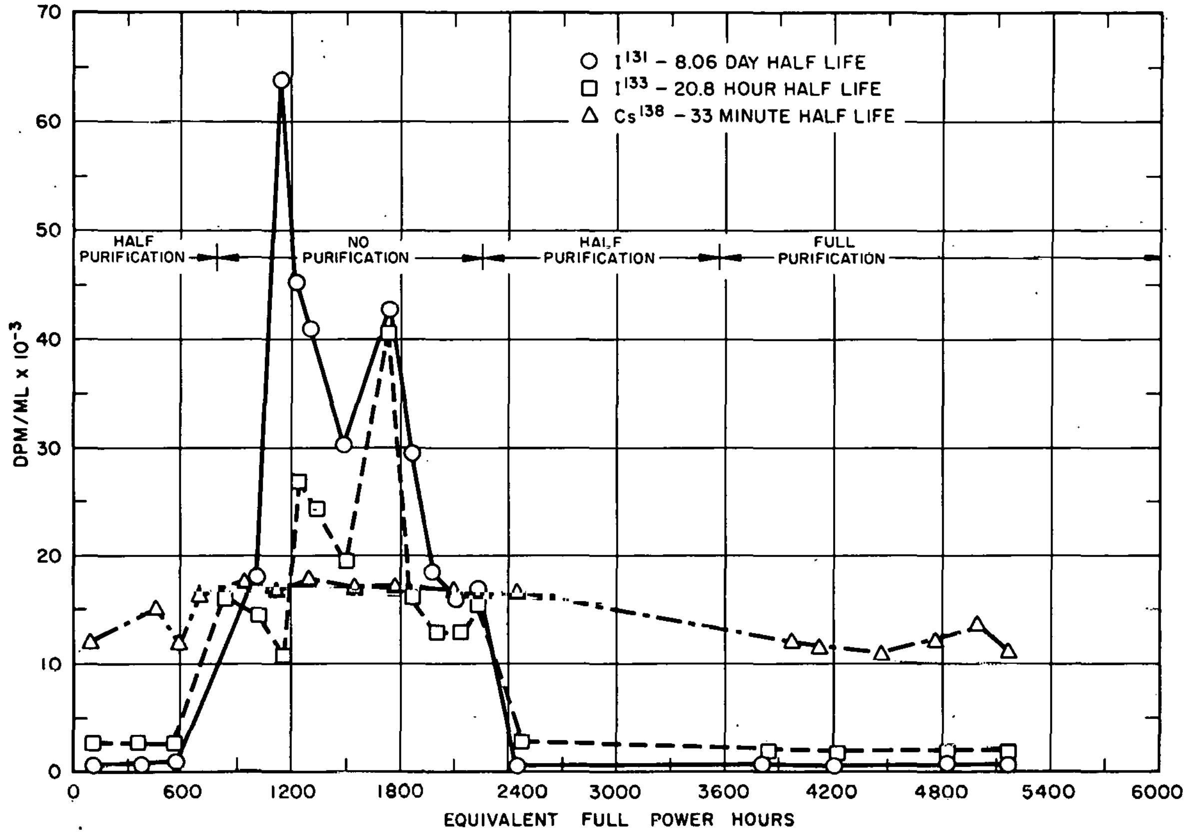


Figure V-6. PWR I^{131}, I^{133}, and Cs^{138} Activity Levels -- Ion Exchange Influent.

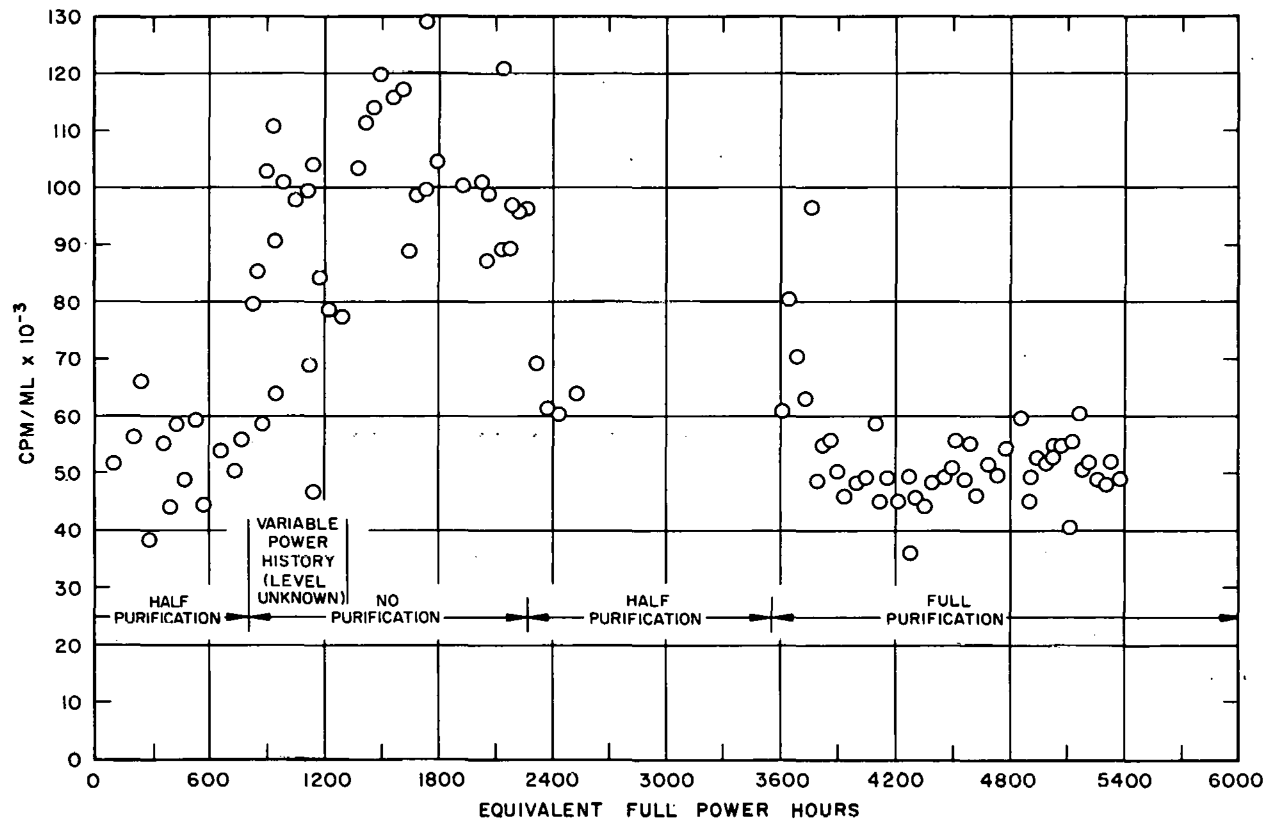


Figure V-7. PWR 15-Minute Gross Nonvolatile Gamma Activity at 100 Percent Power.

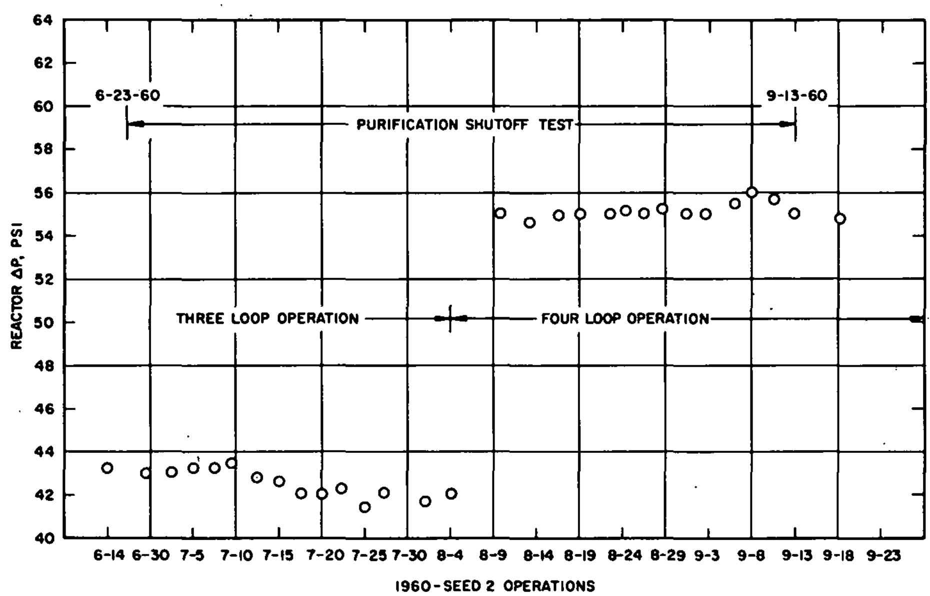


Figure V-8. PWR Reactor Vessel Pressure Drop (As Read).

TABLE V-I

CORROSION PRODUCT ACTIVITY IN THE PRIMARY COOLANT*

(dpm/ml)

Core I EFPH	Co^{60}		Co^{58}		Mn^{54}		Fe^{59}		Cr^{51}		Hf^{181}		Zr^{95}	
	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud
Seed 1														
500	1	-	1	-	-	-	0.5	-	4	-	0.03	-	0.02	-
1000	6	-	1	-	.2	-	6	-	-	-	0.5	-	0.3	-
1500	13	-	2	-	.2	-	5	-	-	-	0.4	-	0.4	-
2000	36	-	7	-	-	-	10	-	-	-	0.4	-	0.2	-
2500	29	-	5	-	-	-	25	-	-	-	†	-	†	-
2900	Beginning of half-purification run													
3235	130	307	11	47	9	39	12	36	†	8	†	6	†	23
3336	-	22	-	6	-	2	-	6	-	6	-	0.2	-	0.5
3558	-	114	-	33	-	13	-	28	-	40	-	1	-	1
4549	-	39	-	-	-	3	-	-	-	-	-	-	-	-
4716	8	50	-	-	8	6	-	-	-	-	-	-	-	-
4882	-	22	-	-	-	2	-	-	-	-	-	-	-	-
4900	System returned to full-purification													
5532	-	26	-	-	-	3	-	2	-	3	-	0.7	-	0.4
5786	7	32	2	5	16	3	33	5	†	6	†	3	†	1
5806	End of Seed 1 Life - October 7, 1959													
Seed 2	Initial criticality for Seed 2 achieved on May 6, 1960 - System on half-purification													
6525	12	139	†	†	18	1.3	2	1.7	8	72.5	2	5.1	†	†
6576	Beginning of Purification Shutoff Test													
6599	‡	690	‡	‡	‡	80	‡	115	‡	126	‡	30	‡	4.9
6991	‡	217	‡	17	‡	18	‡	24	‡	15	‡	8	‡	3.3
7553	‡	72	‡	10	‡	8	‡	10	‡	3	‡	5	‡	0.4
7953	43	31	5	7	35	35	25	35	18	43	†	7	†	†
8053	System returned to half purification													
9100	‡	40	‡	6	‡	4	‡	3	‡	1	‡	8	‡	0.4
9300	System returned to full purification													
9370	‡	268	‡	55	‡	18	‡	11	‡	11	‡	45	‡	2.4
10083	11	14	2	1	6	1	9	1	†	2	†	ND†	†	0.1
10865	‡	291	‡	108	‡	14	‡	18	‡	35	‡	51	‡	2.9

*(In dpm/ml. For conversion to the microcurie unit multiply the tabular valve by 4.5×10^{-7}

† Not detected.

‡ Not analyzed.

TABLE V-J

CORROSION PRODUCT ACTIVITY IN THE PRIMARY COOLANT

($\mu c/ml \times 10^{-6}$)

Core I EFPH	Co^{60}		Co^{58}		Mn^{54}		Fe^{59}		Cr^{51}		Hf^{181}		Zr^{95}	
	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud	Filtrate	Crud
500	0.46	-	0.46	-	-	-	0.23	-	1.82	-	0.014	-	0.009	-
1000	2.76	-	0.46	-	0.09	-	2.76	-	-	-	0.23	-	0.14	-
1500	5.92	-	0.92	-	0.09	-	2.28	-	-	-	0.18	-	0.18	-
2000	16.4	-	3.22	-	-	-	4.55	-	-	-	0.18	-	0.09	-
2500	13.2	-	2.30	-	-	-	11.4	-	-	-	-	-	-	-
2900	Beginning of half purification run													
3235	59.2	140	5.0	21.4	4.1	17.7	5.46	16.4	-	3.68	-	2.76	-	10.5
3336	-	10	-	2.76	-	0.92	-	2.76	-	2.76	-	0.092	-	0.23
3558	-	51.8	-	15	-	5.92	-	12.7	-	18.4	-	0.46	-	0.46
4549	-	17.7		-	-	1.38	-	-	-	-	-	-	-	-
4716	3.68	23.0	-	-	3.68	2.76	-	-	-	-	-	-	-	-
4882	-	10	-	-	-	0.92	-	-	-	-	-	-	-	-
4900	System returned to full purification													
5532	-	11.8	-	-	-	1.38	-	0.92	-	1.38	-	0.32	-	0.18
5786	3.22	14.5	0.92	2.30	7.28	1.38	1.38	2.30	-	2.72	-	1.38	-	0.46
5806	End of Seed 1 Life - October 7, 1959													
Seed 2	Initial Criticality for Seed 2 achieved on May 6, 1960 - System on half purification													
6525	5.46	63.2	-	-	8.2	0.59	0.92	0.77	3.68	33	0.92	2.32	-	-
6576	Beginning of Purification Shutoff Test													
6599	-	313	-	-	-	36.8	-	52.8	-	58	-	13.8	-	2.25
6991	-	99	-	7.82	-	8.27	-	11.0	-	6.9	-	3.68	-	3.82
7553	-	33.2	-	4.6	-	3.68	-	4.6	-	1.38	-	2.3	-	0.18
7953	19.8	14.3	2.30	3.22	16.1	16.1	-	16.1	-	19.8	-	3.22	-	-
8053	System returned to half purification													
9100	-	18.4	-	2.76	-	1.84	-	1.38	-	0.46	-	1.38	-	0.18
9300	System returned to full purification													
9370	-	122	-	25.2	-	8.28	-	5.05	-	5.05	-	20.6	-	1.1
10083	5.05	6.45	0.92	0.46	2.76	0.46	4.15	0.46	-	0.92	-	-	-	0.046
10865	-	133	-	49.6	-	6.45	-	8.28	-	16.1	-	24.2	-	1.33

CHAPTER 2

FAILED ELEMENT DETECTION AND LOCATION SYSTEM (FEDAL)

The PWR Core 1 blanket is comprised of 113 fuel assemblies of Zircaloy clad UO_2 elements. Samples of the effluent from each blanket fuel assembly are taken continuously. A multiport valve at the top of the reactor vessel selects the sample of effuent from a pair of fuel assemblies and directs them to a pair of monitors. The flows from all other fuel assemblies are bypassed and returned to the system. The monitors consist of a coil of stainless-steel pipe suspended in a tank of water. BF_3 tubes in the water tank detect the presence of neutrons from 54s Br^{87} and 22s I^{137} fission product delayed neutron emitters. Delayed neutron activity from the 4s N^{17}, formed by the reaction $O^{17} + n \rightarrow N^{17} + p$, does not interfere since it effectively decays during the delay time to the monitor.

The measure of the FEDAL systems ability to locate a defect is the signal-to-background ratio. This is defined as the ratio of the activity in the effluent from any assembly containing a defected fuel element to the activity in the effluent of an equivalent assembly without a defect. In general the background activity comes from three sources:

1. From the fissioning of U^{235} and the Pu^{239}, and Pu^{241} formed by the neutron capture of U^{238} in natural uranium found as an impurity in core Zircaloy,
2. From the recirculated delayed neutron activity from any failure in the core,
3. Cosmic rays and tube noise.

The presence of relatively high levels of natural uranium contamination in core Zircaloy is particularly bad. Fissionable Pu^{239} and Pu^{241} form continuously with irradiation, thus increasing the delayed neutron activity from this source and making the ability to locate a failure more difficult with time. It was estimated, from the levels of fission products in the reactor coolant, that Core 1 Seed 1 had the equivalent of 2.54 ppm of natural uranium in core Zircaloy.

Because of the relatively high background activity from U^n contamination it was anticipated that under steady-state conditions, a small hole in the cladding of a UO_2 rod could probably not be located although more serious defects could be located. However, it is characteristic of a UO_2 compact with a small hole in the cladding that the fission product activity in the reactor coolant peaks during reactor start-ups. This is frequently referred to as "waterlogging". It is postulated that water enters the defect during shutdown periods and leaches the soluble fission products from the fuel surface; this water, with a very high specific activity, is then expelled from the fuel when the fuel temperature reaches the boiling point during the subsequent reactor start-up operation. By holding the multiport valve on various ports during reactor start-ups, strong peaking of delayed neutron activity was observed on port 53 monitor 2 during Seed 1 operation. The activity on monitor 1 showed only a gradual increase with reactor power (Figure V-9). This was taken as an indication that a defected UO_2 rod existed in what at that time was believed to be assembly F-2. Peaking of delayed neutron activity during reactor start-ups was not observed on any other ports investigated during Seed 1 operations. During Seed 1 refueling, assembly F-2 was removed and replaced.

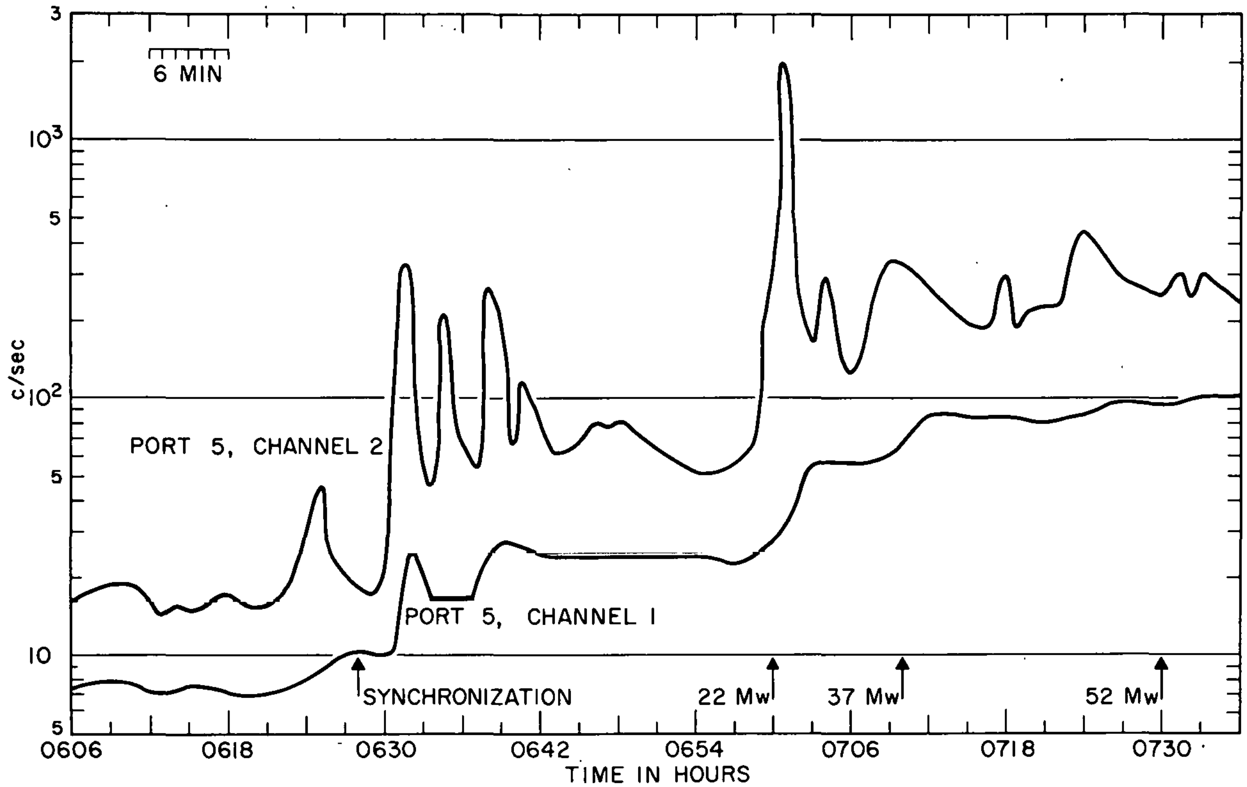


Figure V-9. Delayed Neutron Activity in PWR Following Reactor Start-up on September 22, 1958.

On May 15, 1960, early in Seed 2 operations, the multiport valve was again held on port 53 during a reactor start-up. Peaking of the delayed neutron activity occurred again on monitor 2. There appeared to be only two possible explanations for this second activity peak: (1) The unlikely possibility that the replacement assembly for F-2 had a defected fuel element when it was installed in the reactor, or (2) the index relating the various assemblies to the ports on the multiport valve was in error.

Throughout Seed 1 and Seed 2 operation, the relative delayed neutron activity (defined as the ratio of the activity in the effluent from a given assembly to the average activity from all assemblies) from each assembly had been constant, with an average standard deviation of only about 5 percent. The first measurement performed during Seed 2 operation, June 6, 1960, showed that the relative activity on port 5 monitor 2 had decreased from a Seed 1 average of 0.96 to 0.44. The index indicated this was apparently assembly J-5. This decrease was well outside the statistical variations for the entire core. A smaller, but also statistically significant, decrease was also notes on port 55 monitor 2 (indicated assembly E-2). It would be expected that a new assembly installed during Seed 1 refueling would show such a decrease since the U^{238} impurity in its Zircaloy would not have grown into fissionable Pu^{239} or Pu^{241}. These apparent decreases in J-5 and E-2 activities were unexpected, since these assemblies had not been disturbed during refueling. However, fuel assembly F-2 was completely replaced; the third fuel bundle from the bottom, the blanket shell, and the FEDAL sample tube were replaced in assembly E-6 during Seed 1 refueling. This led to the suggestion by the Duquesne Light Company that the multiport valve was actually traveling opposite to the intended direction, i.e., the actual port number would equal 58 minus the indicated port number. Therefore, port 5 monitor 2 (indicated by the index as assembly J-5) and port 55 monitor 2 (indicated by the index as assembly E-2) would actually be assemblies F-2 and E-6 respectively which were expected to show a decrease in the relative delayed neutron activity.

Further evidence that the valve was traveling in the reverse direction was indicated by the fact that the values of the relative delayed neutron activities showed a completely random distribution with regards to their apparent location in the various regions of the core as indicated by the index. However, when the relative activities were plotted using the index corrected for reversed travel, the distribution was more what one would expect, i.e., the distribtion was relatively symmetrical about the core and high values of the relative activities were in the high neutron flux regions and vice versa. Figure V-10 shows such a plot of the average relative delayed neutron activities and their standard deviations for Seed 1 and 2.

Subsequently, the electrical leads to the multiport valve were reversed to reverse the direction of valve travel. Confirmation that the direction of travel of the valve was reversed was demonstrated by the fact that the relative delayed neutron distribution with regard to the various ports was reversed. However, because the operation of the multiport valve was erratic in the reversed direction, it was returned to the original direction of drive and the index reversed to give the correct relationship between port number and fuel assembly.

During a reactor start-up on July 16, 1960, peaking of delayed neutron activity was observed on port 57 monitor 1. The corrected index indicates that this is assembly K-8. Figure V-11 shows a reproduction of the FEDAL traces during this start-up. The trace from monitor 2 is given to show the normal gradual increase with reactor power. The peaking of the delayed neutron activity from assembly K-8 and J-5 was confirmed several times later in Seed 2 life. Both of these assemblies are scheduled to be replaced during Seed 2 refueling. Subsequent examination of assembly J-5 and K-8 revealed a "pin hole" defect in one rod. This defect was made in fabrication and was not the result of operation.

The delayed neutron activity was observed in the effuent from all blanket fuel assemblies at least once during reactor start-ups during Seed 2. The characteristic peaking of activity from defected UO_2 rods was only observed in assemblies J-5 and K-8. The rates of release of delayed neutrons from these defects under steady-state conditions were not sufficiently great that they could be observed above the background activity. The release of other fission products from these defects was important only in the case of soluble, relatively long-lived fission products I^{131} and Xe^{133}. The levels of all other important fission products in the reactor coolant were controlled as in Seed 1 by 2.54 ppm natural uranium contamination in core Zircaloy.

B LOOP — N — A LOOP

W — E

	2	3	4	5	6	7	8	9	10	11	12	13	14
A													
B				19 0.986	18 0.921	17 0.954	16 0.958	17 0.969	18 0.909	19 0.977			
C			11 0.982	12 1.096	13 1.079	14 1.107	15 1.045	14 1.074	13 1.159	12 0.991	11 0.937		
D		11 1.006	10 1.176								10 1.021	11 0.883	
E	19 0.908	12 1.076			a 7 0.796	8 1.075	9 1.134	8 1.073	7 1.081			12 0.998	19 0.917
F	c 18 0.515	13 1.103		7 1.118	6 0.997	5 0.985	4 1.044	5 1.037	6 1.021	7 1.093		13 0.927	18 0.911
G	17 0.951	14 1.149		8 1.099	5 1.091	3 0.951	2 0.904	3 0.985	5 1.028	8 1.019		14 0.938	17 0.908
H	16 0.966	b 15 1.072		9 1.108	4 1.048	2 0.911	1 0.941	2 0.935	4 1.019	9 1.119		15 0.924	16 0.885
I	17 0.941	14 1.104		8 1.057	5 1.014	3 0.956	2 0.967	3 0.960	5 1.042	8 1.205		14 0.930	17 0.857
J	18 0.960	13 1.123		* 7 1.204	6 1.020	5 0.974	4 1.055	5 1.045	6 1.093	7 1.240		13 0.971	18 0.905
K	19 0.898	12 1.027			7 1.101	8 1.134	* 9 1.128	8 1.107	7 1.202			12 0.969	19 0.868
L		11 0.920	10 1.099								10 1.020	11 0.937	
M			11 0.931	12 0.911	13 0.908	14 0.966	15 0.945	14 0.976	13 0.956	12 0.922	11 0.914		
N				19 0.841	18 0.895	17 0.953	16 0.909	17 0.903	18 0.904	19 0.867			
O													

D LOOP — S — C LOOP

* DEFECT

a THIRD BUNDLE FROM BOTTOM, BLANKET SHELL, AND SAMPLE TUBE REPLACED DURING REFUELING

b THIRD BUNDLE FROM BOTTOM REPLACED

c ASSEMBLY COMPLETELY REPLACED

Figure V-10.. Relative Delayed Neutron Activities in PWR Core I.

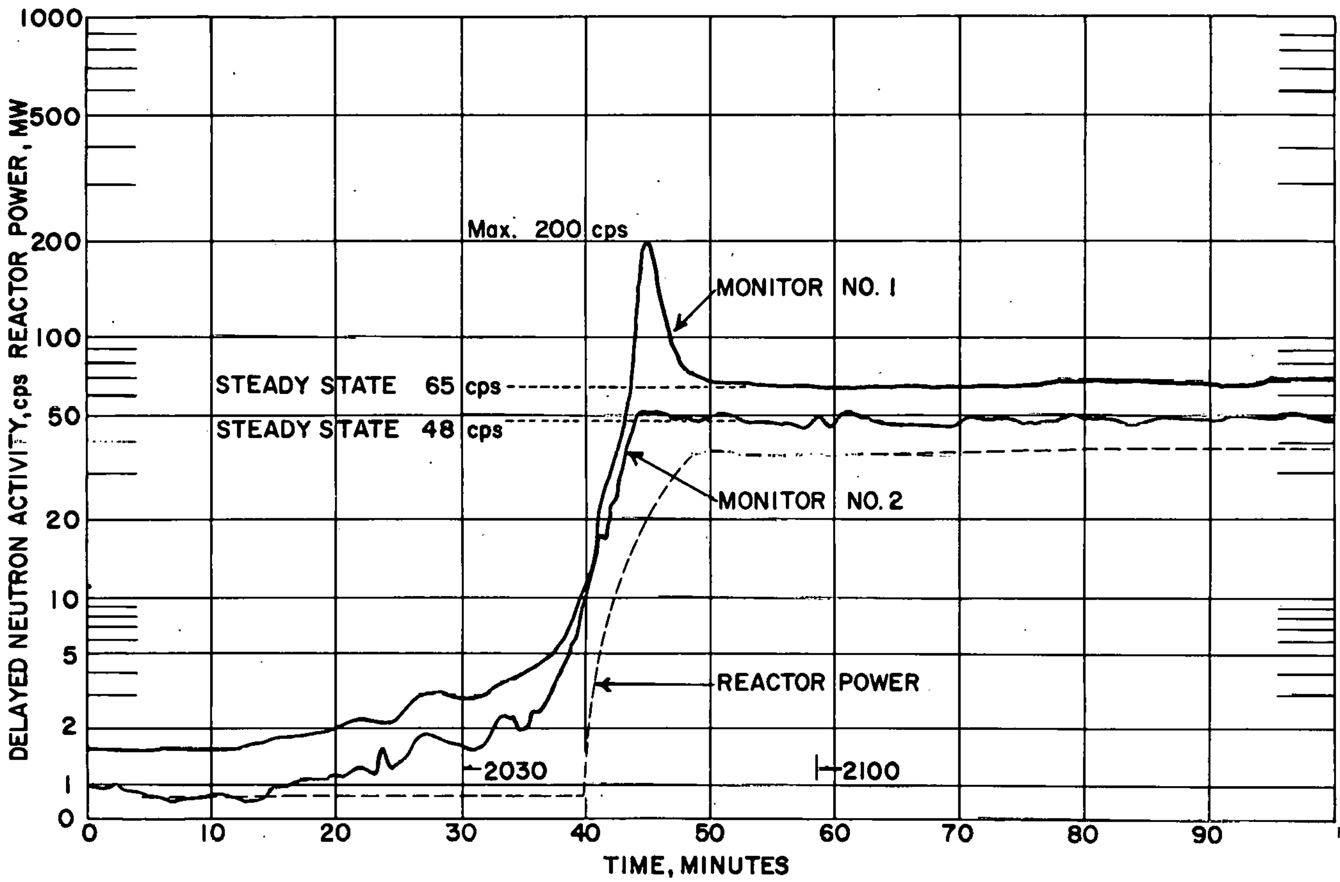


Figure V-11. PWR FEDAL Trace.

CHAPTER 3

RADIATION LEVEL BUILD-UP EXPERIENCE ON COMPONENTS AND PIPE

Introduction

It is important to the operation and maintenance of a nuclear power station to know the extent of radiation levels in the working areas since these levels determine accessibility. It is also important to the designer of a nuclear power station to know both the extent and rate of radiation build-up in components; such data, when compared to nuclear plants operating under coolant chemistry conditions different from those at Shippingport, can aid in the establishment of the best chemical control program for nuclear systems.

Since the beginning of Seed 1 operation, the radiation intensity of reactor coolant and purification system components have been frequently monitored to provide this information. Radiation levels are determined by periodic surveys of selected components which are in contact with the reactor coolant water. These components include piping, various heat exchangers, demineralizers and pumps, as well as test loops specifically installed to provide this data.

Source of Radiation

The source of this long-lived radiation emanating from reactor coolant components is crud which is deposited on metal surfaces. From a chemical standpoint, crud is a product formed as a result of corrosion and wear to the material used to construct the system components and the core itself. This includes such materials as type 304 and 410 stainless steels zirconium, hafnium, inconel, and stellites. The corrosion products are primarily from 304 stainless steel while the wear products are mainly stellite. From a radiochemical standpoint, the crud at Shippingport is composed primarily of cobalt-60 (approximately 70 percent of the total gamma activity during Seed 2) with lesser amounts of cobalt-58, iron-59, manganese-54, chromium-51, hafnium-181, and zirconium-95. These nuclides are formed by an activation process in the reactor flux. A detailed evaluation of the chemical and radiochemical data is presented in a subsequent chapter.

Radiation Levels - Reactor Vessel Head

Portable survey instruments were used to survey the reactor vessel head 13 times during Seed 2 operation. With the reactor subcritical (usually after extended power operation), selected locations on the reactor vessel head were monitored for radiation level approximately 30 hours after shutdown. Where possible, all readings were on contact. Thirty-two control rod drive mechanisms (at one-foot intervals for six feet inside and outside of the trellis), ten fuel ports (top, middle, and bottom positions), the multiport valve, three flow measurement instrumentation enclosures, seed exit thermocouple junction box, and seed metal thermocouple junction boxes (all at top, middle and bottom positions) were surveyed. The highest radiation reading was 1000 mr/hr and occurred at a location of one foot above the reactor vessel head on rod drive mechanism K-11. The average of all mechanisms during this same survey was 700 mr/hr at the one foot level and 180 mr/hr at the six foot level with an average value of approximately 400 mr/hr.

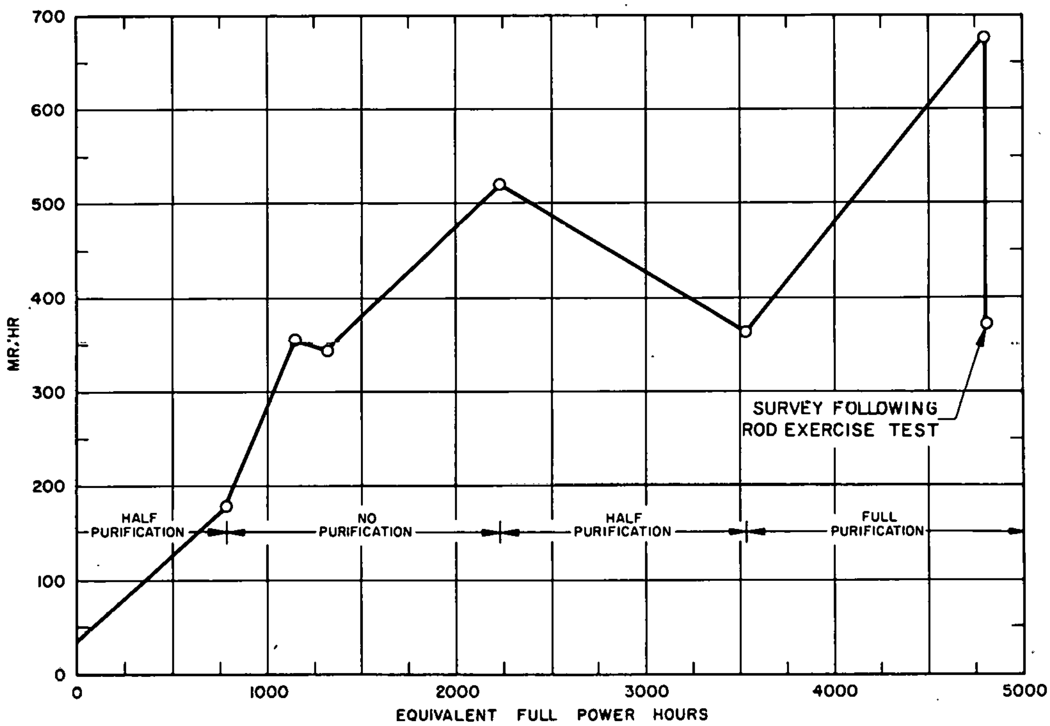


Figure V-12. Long-Lived Radiation Levels of Mechanisms (average of approximately 75 points taken in contact with the mechanisms).

During the period of 1475 EFPH when the reactor coolant system was operated without purification flow, the reactor vessel head was surveyed four times. The K-11 control rod mechanism showed increased in activity buildup with each successive survey, from 400 to 1000 mr/hr. However, all survey points on the head did not exhibit this same rate of activity buildup; some points showed little, if any, tendency to increase.

It was observed from an evaluation of the head survey data that the control rod mechanisms radiation levels fluctuated greatly from one survey to another and the variation seemed to be related to the extent of rod motion. This is graphically presented in Figure V-12. As a result, two reactor vessel head surveys were taken; one before and one following a control rod scram test. Each control rod was individually raised to the upper program limit and intentionally scrammed. The head survey following this scram test showed a 47 percent decrease in the radiation activity of the control rod mechanisms at a level of one foot. It was postulated that the action of the control rod lead screw during withdrawal and the turbulence created during the scram may have loosened crud adhering to the inner surface of the mechanism housings.

A second test was performed to determine the effectiveness of a scram versus simple rod motion in reducing the radiation levels of the mechanisms. A survey was made following a period of operation to establish initial radiation levels. The rods in the north and eastbank were then scrammed from the maximum heights (2 scrams from a 5-inch height were also performed) and the rods in the south and west banks were exercised by withdrawing and inserting each rod for two full travel cycles. After a second radiation survey, the rods in the south and west bank were scrammed and the rods in the other two banks were exercised. The following observations were made from the survey data:

North and east banks.

1. After the scram, the average radiation level at the 1, 2, and 3-foot levels was reduced 20 percent from an average level of 556 to 445 mr/hr.
2. After two rod exercises, the average radiation level at the 1, 2, and 3-foot levels was 453 mr/hr. A total reduction of 19 percent was thus obtained.

South and west banks.

1. After the two rod exercises, the average radiation levels at the 1, 2, and 3-foot levels was reduced 16 percent from an average level of 582 to 487 mr/hr.
2. After a scram, the average radiation level at the 1, 2, and 3-foot levels was 437 mr/hr. A total reduction of 25 percent was thus obtained.

Four full travel rod exercises were then performed since it was not known if the maximum reduction in radiation levels had been achieved. The results of a third survey showed that the average radiation level of 445 mr/hr for all rods at the 1, 2, and 3-foot levels had increased to 540 mr/hr. The average radiation levels for the 4, 5, and 6-foot levels and decreased from 155 to 128 mr/hr. As a result of this increase for the 1, 2, and 3-foot levels, all reactor vessel head survey data for Seed 2 was reviewed and the following conclusions were drawn:

1. Rod exercises will reduce radiation levels in the higher elevation and move crud to lower elevations.
2. Scrams are effective in reducing radiation levels in the lower elevations.
3. Most effective combination appears to be rod exercise followed by scramming the rods from their full height.

Another observation made during Seed 2 operation involves the effect of a reactor plant cooldown on measured activities. A decrease in the amount of deposited crud could be caused by a cooldown due to the difference in the coefficient of expansion of the crud and metal surface which causes the crud to loosen or flake off the metal surface and be transported away by the circulating reactor coolant. This is true not only for reactor vessel head surveys but also for reactor coolant loop piping and components, as well as for the purification system surveys.

Reactor Coolant Piping and Components

Piping Surveys

The reactor coolant loop piping was surveyed eight times during Seed 2 operation following an extended period of operation. (See Figures V-13 through V-20.) Portable survey instruments were used each time and the survey usually took place following an extended power run. For each survey the locations were identified by the presence of metal cans strapped to the coolant loop piping and to the various components such as the reactor coolant pump volute, inlet and outlet of the heat exchangers, and coolant loop valves. The probe of the survey instrument is inserted into the can and rotated until the highest reading is obtained. The cans are used to insure that the probe position is duplicated in each survey.

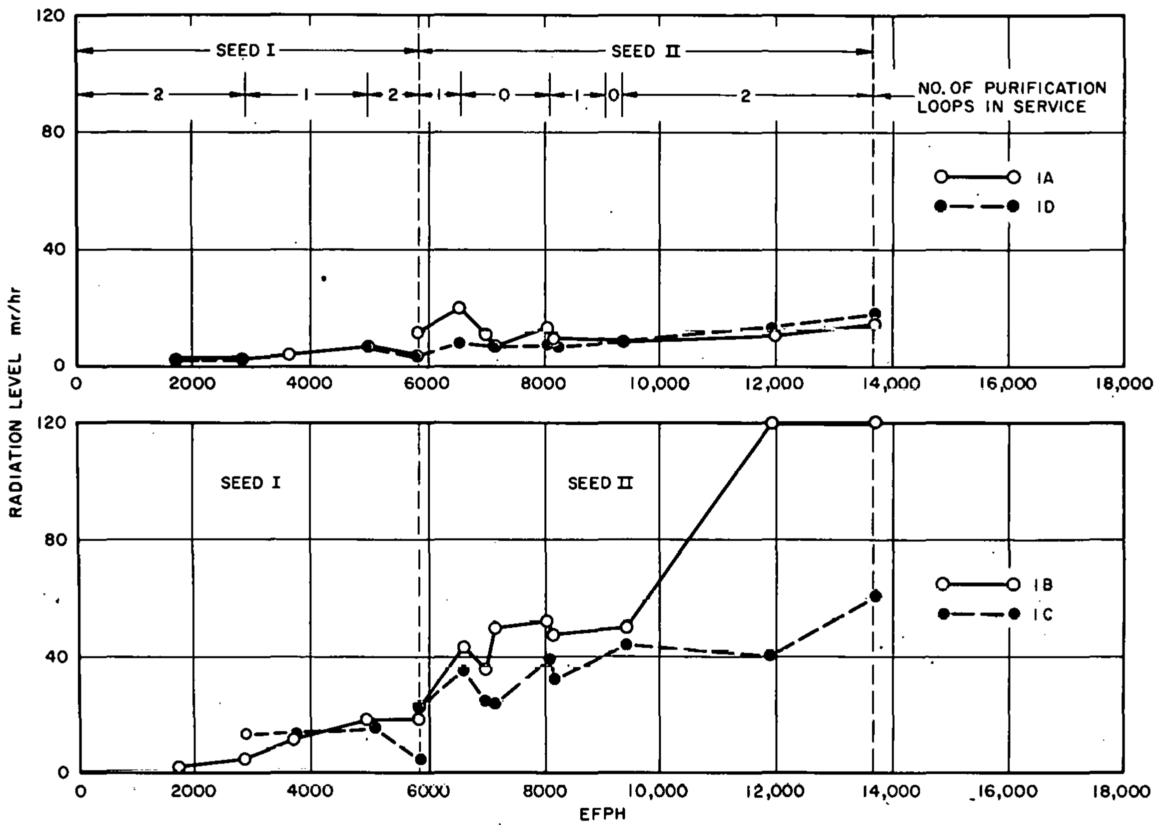


Figure V-13. Radiation Survey -- Main Check Valve.

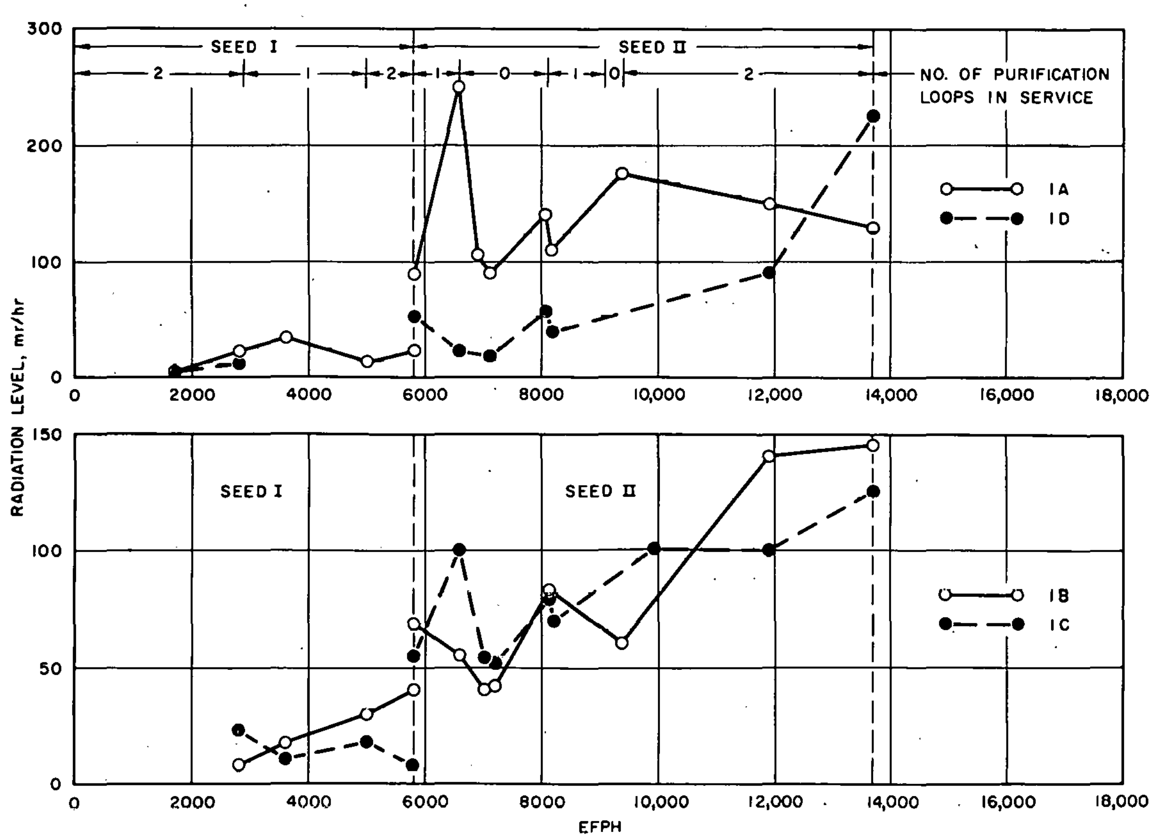


Figure V-14. Radiation Survey -- Heat Exchanger Inlet, Top.

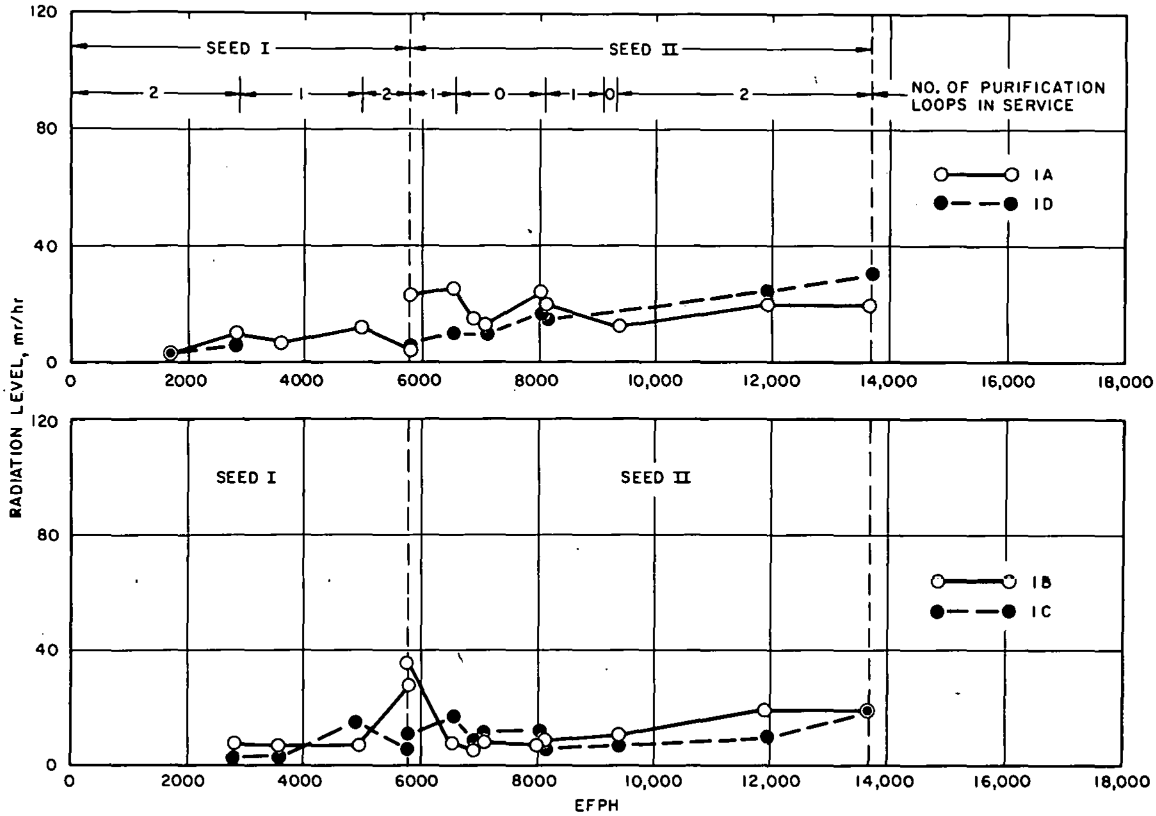


Figure V-15. Radiation Survey -- Heat Exchanger Bottom Casing.

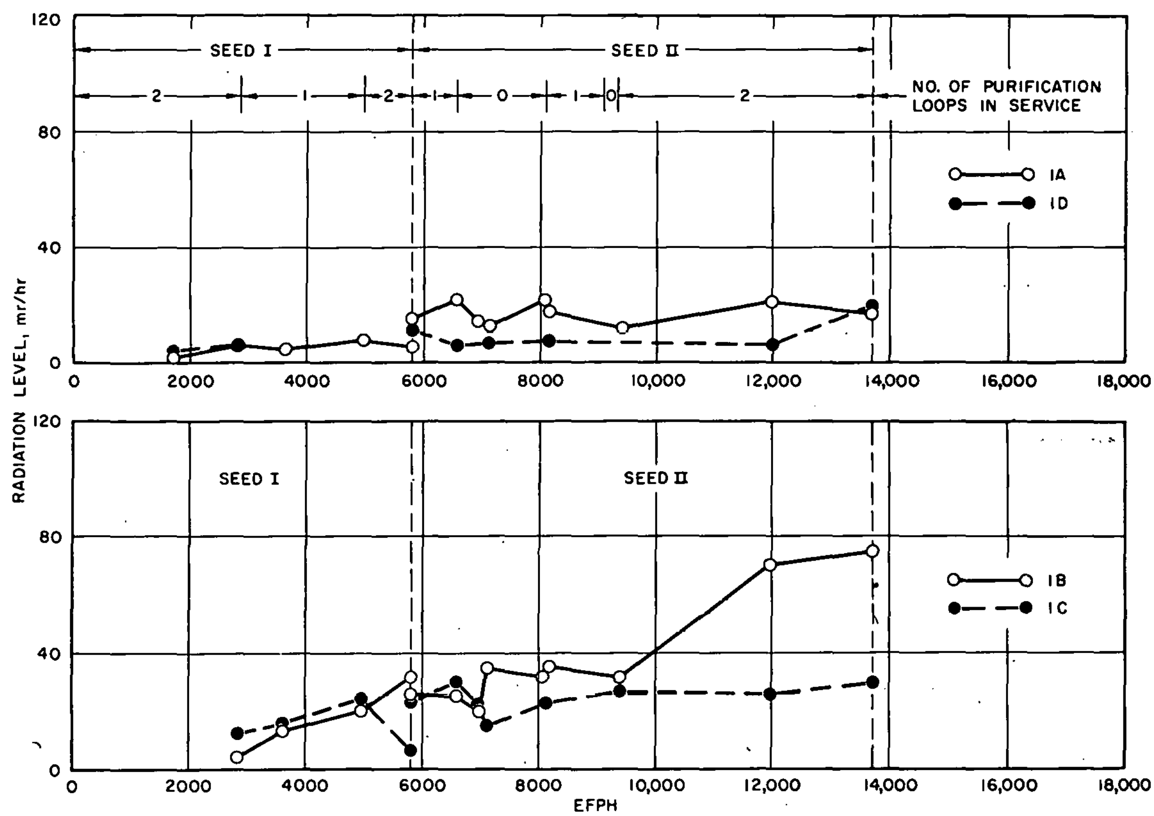


Figure V-16. Radiation Survey -- Heat Exchanger Outlet, Top.

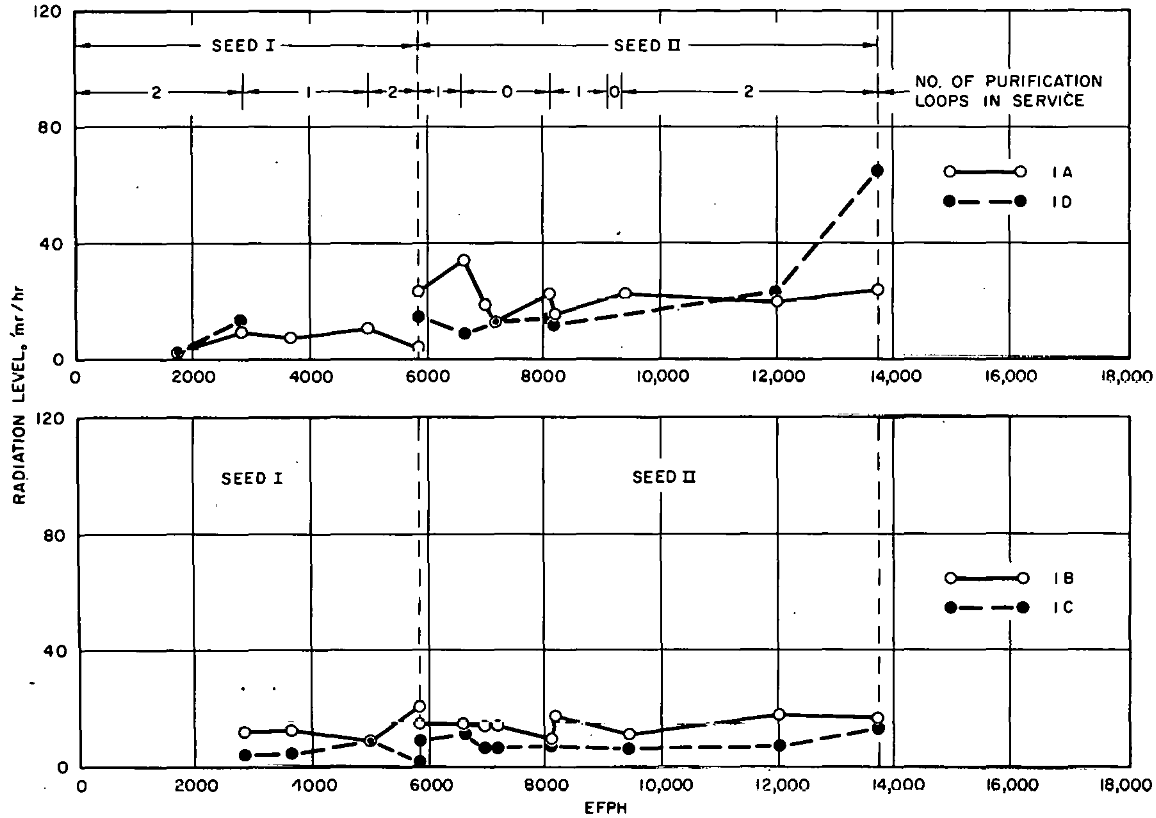


Figure V-17. Radiation Survey -- Coolant Pump Volute.

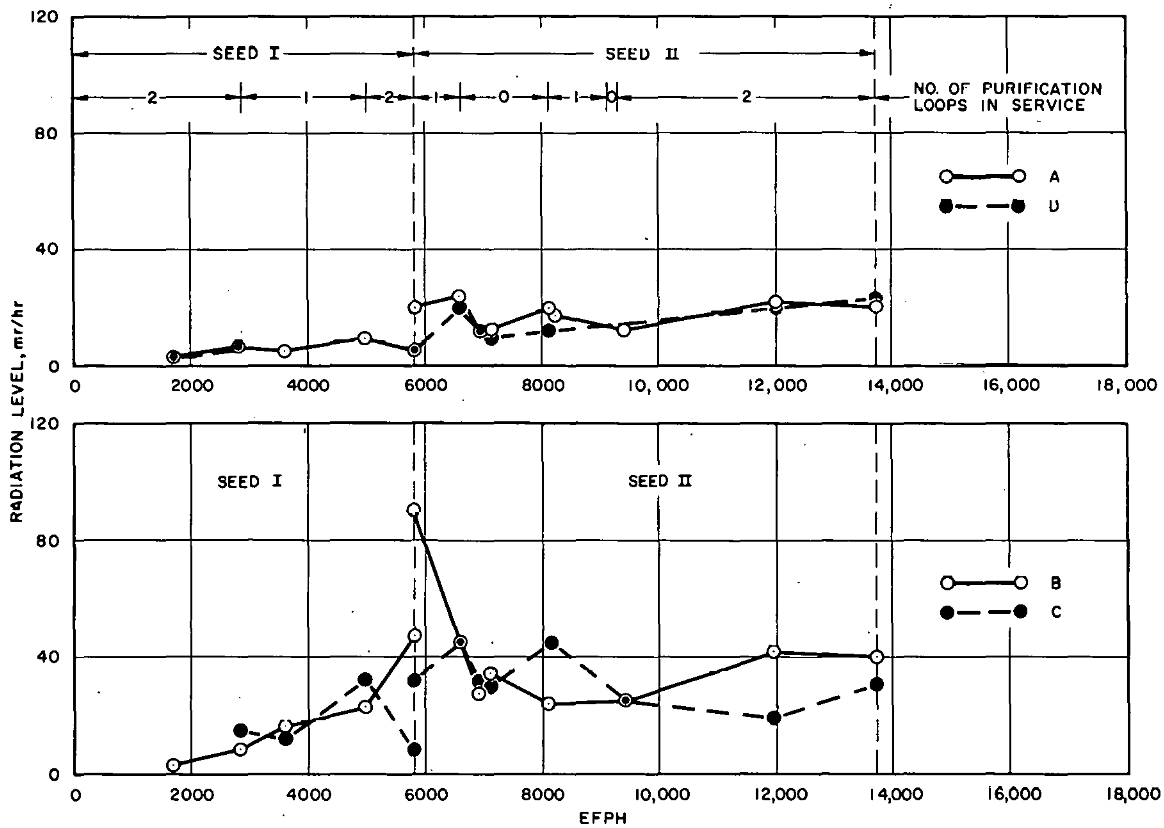


Figure V-18. Radiation Survey -- 18 Inch M.O. Stop Valve Bottom Outlet, Cold Leg.

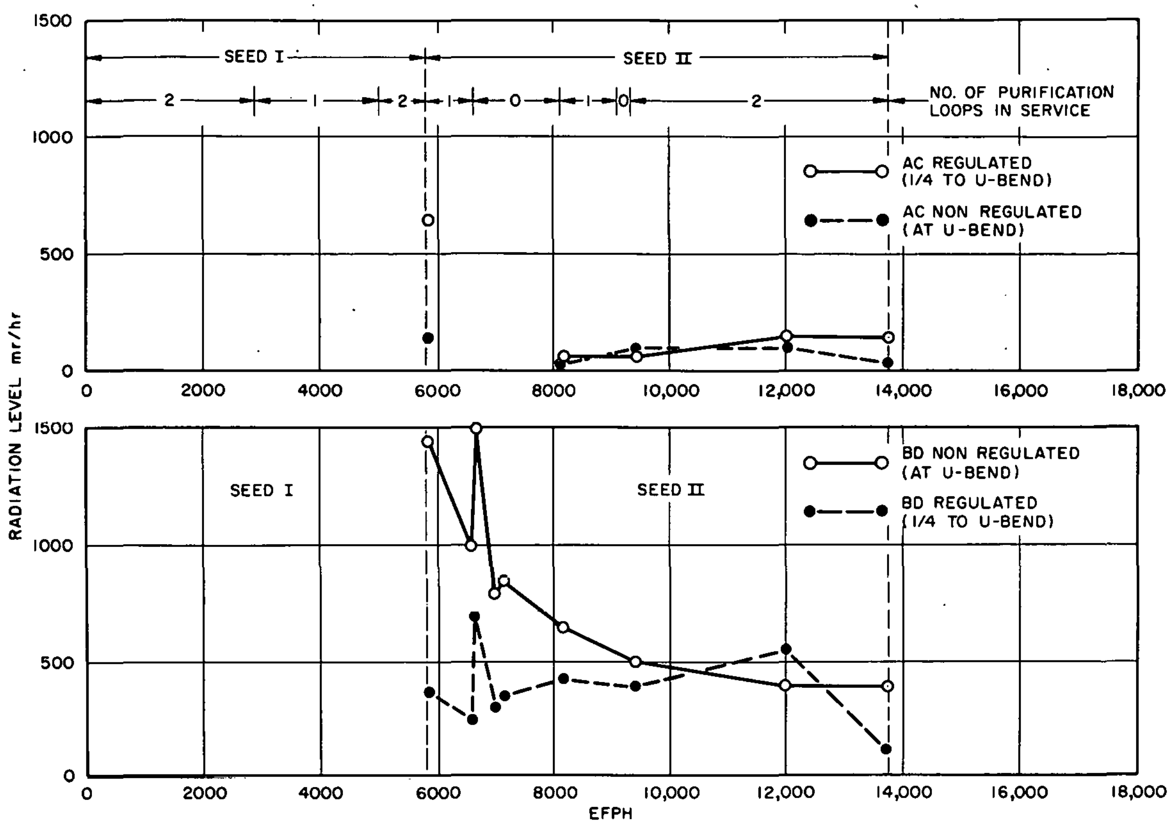


Figure V-19. Radiation Survey -- Purification System.

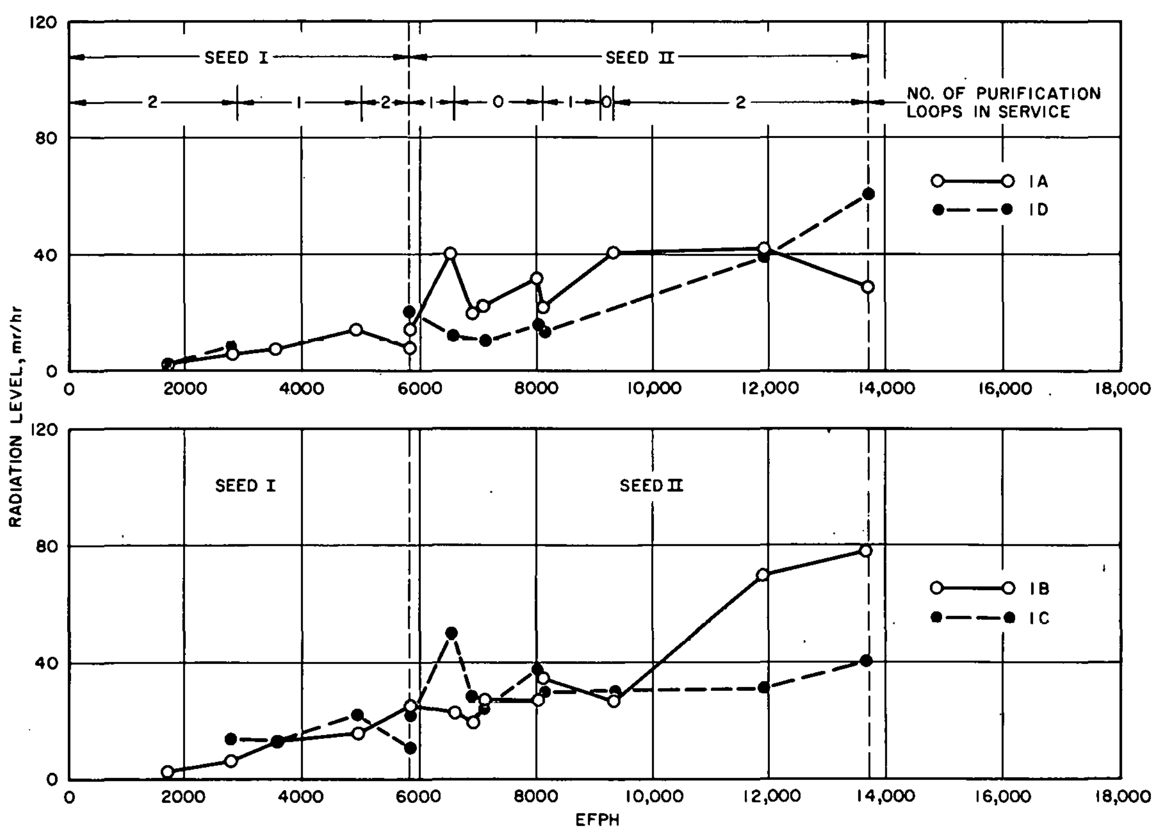


Figure V-20. Radiation Survey -- 18 Inch M.O. Stop Valve Bottom Inlet, Hot Leg.

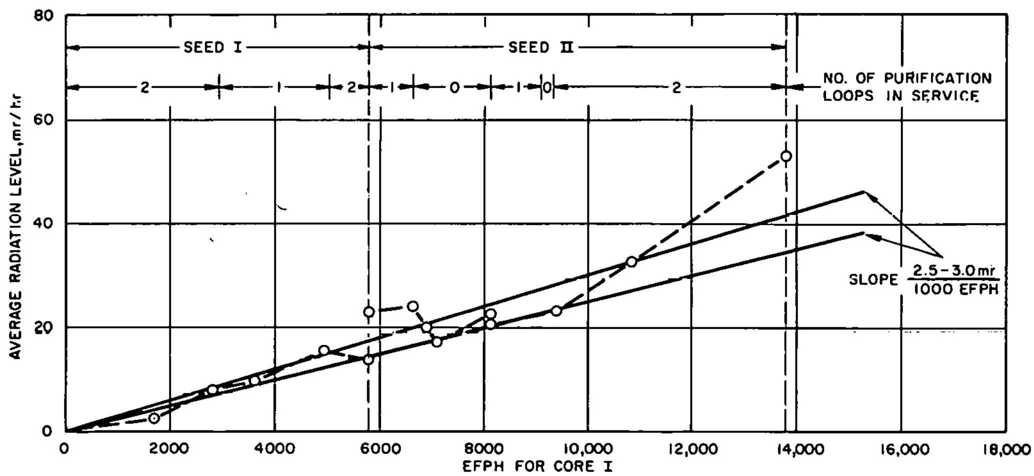


Figure V-21. Average Radiation Level for Reactor Coolant System.

Generally the pattern of activity was the same for the 1A and 1D coolant loop, which are of identical construction; the same holds true for the 1B and 1C coolant loops. For the 1A and 1D loops, the highest activity was measured on the top of the heat exchanger inlet. The highest activity measured was 250 mr/hr for the 1A coolant loop and 100 mr/hr for the 1D loop. In contrast, the highest activity for the 1B and 1C coolant loops was located on the hot leg of the piping at the interconnecting duct to the reactor chamber. The highest activity for the 1B loop was 410 mr/hr and 275 mr/hr for the 1C loop at the same location.

In order to evaluate the extent of radiation build-up in the reactor coolant system, the test data for each survey (as presented in Figures V-13-20) were averaged and the average results are shown in Figure V-21. Between 24 and 32 observations were used for each average. Using 11 of the 14 points, the data would indicate a rate of radiation build-up of 2.5 to 3.0 mr/hr per 1000 EFPH. Of the three points not used, two were based on data taken during the initial operating period for Seed 2. It can be postulated that these higher readings were due to a redistribution of crud disturbed by the refueling operation. The significance of the data from the last survey at 13,700 EFPH cannot be evaluated until data is obtained for Seed 3.

Component Survey

During Seed 2 operation, the 1A reactor coolant loop was removed from service due to leaking heat exchanger tubes. The handhole covers were removed from both the inlet and outlets ends and a radiation survey was made. Portable survey meters and film badges were used to determine the radiation intensity inside the hemispherical head of the heat exchanger inlet end. The results of this survey revealed:

1. A high radiation (20 R/hr maximum) intensity in this area, and
2. That consistent data was obtained between the portable survey instruments and the film badges.

The following data show the results of the survey:

Portable survey instrument.

On contact at center of tube sheet - 20 R/hr B-γ;
6 inches from tube sheet - 13 R/hr B-γ;
12 inches from tube sheet - 10 R/hr B-γ.

Film badge data.

3/4-inch from tube sheet	- unshielded	- 20.1 R/hr
	- beta shielded	- 18.6 R/hr
6 inches from tube sheet	- unshielded	- 16.8 R/hr
	- beta shielded	- 13.5 R/hr

The film badge showed that at least two-thirds of the activity was coming from crud deposited on the face of the tube sheet. Based on radiochemical data, more than 50 percent of the gamma activity is due to cobalt-60.

Heat Exchangers

The 1BD regenerative and nonregenerative heat exchangers were surveyed seven times during Seed 2 operation. The results are shown in Figure V-13. Due to the inaccessibility of these heat exchangers during Seed 1, they were not surveyed at the same locations, and, therefore, no comparison can be made between Seed 1 and Seed 2 operation.

Certain locations, especially on the 1BD regenerative heat exchanger, indicated a high radiation field. Just prior to Seed 2 operation, at a distance three-quaters of the way from the inlet to the U-bend, an activity of 1450 mr/hr was measured. This same point reached a maximum of 1800 mr/hr after 773 equivalent full power hours of Seed 2 operation. Subsequent surveys have shown a steady decline in activity. The 1AC heat exchangers, isolated during some of Seed 2 operation, were surveyed less frequently but the data also indicated a considerably lower activity field than for the 1BD heat exchangers.

The regenerative heat exchanger of both purification loops consistently showed higher activity than the nonregenerative. This would be expected since both the shell and tube side of the regenerative heat exchanger contain reactor coolant water while the shell side of the nonregenerative heat exchanger contains nonactive component cooling water. Thus the regenerative heat exchanger has a greater surface area exposed to the reactor coolant.

Hairpin Loop

The hairpin loop section of the purification system is a horizontal and vertical section of piping upstream of the demineralizer, which is used to monitor the accumulation of activity on the piping. This piping is surveyed using portable instruments, and also the two legs are physically removed for chemical analysis.

The 1BD hairpin loop was surveyed twice during Seed 2 operation and was removed twice and new sections of piping installed. The 1AC hairpin loop was neither surveyed nor removed during Seed 2.

Mounted on each leg of the hairpin loop is a lead shielded probe holder through which pass the horizontal and vertical piping. During the survey, the probe of the portable survey instrument is inserted through holes in concrete shield doors and into the probe holder. The probe is then rotated until a maximum reading is obtained.

After a section of the horizontal and vertical legs are cut out, the loose crud is brushed out and a radiochemical analysis is performed. The 1BD hairpin loop was in service for the first 760 equivalent full power hours of Seed 2 life. During this period, the 1BD purification system demineralizer was also in service. At the end of this period of operation, the 1BD hairpin loop was isolated and surveyed. Both the horizontal and vertical leg measured 40 mr/hr. The resin was then discharged from the 1BD demineralizer and the plant operated for approximately 1475 equivalent full power hours with flow through the emptied 1BD demineralizer while the 1AC demineralizer was isolated. During this period, the 1BD hairpin loop was periodically isolated and surveyed. The results of those surveys performed 15 hours after isolation are as follows:

EFPH with No Purification	Horizontal Leg (mr/hr)	Vertical Leg (mr/hr)
282.2	35	30
588.4	40	30
941.5	12	10
1192.3	28	35
1392.6	28	23

It can be seen from the preceding table that the effect of no coolant purification on activity buildup was negligible with regard to the hairpin loop.

Coolant Purification System Demineralizers

The 1BD demineralizer of the coolant purification system was surveyed four times during Seed 2 operation while the 1AC demineralizer was surveyed three times. A portable survey instrument with an extra long probe lead was used to survey the demineralizers. The survey probe was lowered into the concrete demineralizer enclosure and activity readings were taken at one-foot intervals along the side of the demineralizer. The resin of both the 1AC and 1BD demineralizers was removed during Seed 2 and new resin charged into each demineralizer. This was the first time that this resin had been replaced since the demineralizers had received their initial charge prior to Seed 1 operation. The following table shows the results of the 1BD demineralizer surveys during Seed 2.

1BD Demineralizer

Date	Resin Service Hours	Max. Level (mr/hr)
6/11/60	11,000	650
11/26/60	458	60
8/18/61	6,126	235

The 1AC demineralizer was not surveyed during Seed 2 operation prior to resin removal. The previous activity survey occurred at the end of Seed 1 life and recorded a maximum of 1100 mr/hr. The following table shows the results of the 1AC demineralizer surveys during Seed 2.

1AC Demineralizer

Date	Resin Service Hours	Max. Level (mr/hr)
11/26/60	208	45
4/3/61	2,766	175
8/18/61	5,262	300

The maximum activity in most cases is located 1-1/2 to 2 feet down from the top of the demineralizer. (Note: The demineralizers are completely filled with resin when loaded, but the bed is compacted during operation). The lone exception to this location was the survey of the 1AC demineralizer after the new resin had been charged, when the maximum activity measured was near the bottom of the demineralizer opposite the purification system outlet line.

Summary

Based on test survey data, the external radiation levels of reactor coolant piping increased at a rate of 2.5 to 3.0 mr/hr per 1000 EFPH when averaged over the Seed 1 and Seed 2 operating periods. These levels are not significant in terms of limiting access to components for operating reasons; however, they do require consideration from a maintenance viewpoint when it is necessary to open the system.

The radiation surveys which were made during Seed 2 operation are planned for continuation during Seed 3. As data becomes available from other pressurized water reactors, comparisons can be made to determine the effectiveness of various types of reactor coolant chemical treatment in limiting the buildup of radioactive crud in system components.

CHAPTER 4

OPERATIONAL CHEMISTRY - TURBINE PLANT

Introduction

Turbine plant chemistry is that associated with a convential station. It includes boiler water chemistry and control, condensate and boiler feed water monitoring, and the water treating system. Boiler water chemical treating is based on the coordinated phosphate - pH method with sulfite as the oxygen scavenger. Chemical treatment is not used to control corrosion in the condensate and boiler feed systems. The water treating systems are convential, utilizing coagulation, filtration, chlorination, softening, and demineralization processes to provide water of the necessary quality to meet all operational requirements for entire station.

The secondary water problems and boiler heat exchanger leakage experienced during Core I Seed 1 operation have been discussed in previous reports (References 1 and 2). This report covers the experience from the layup of the boilers for Seed 1 - Seed 2 refueling to the end of Seed 2 life. Special problems discussed are failure of the air ejector after condenser tubes, failure of the 1A feedwater heater tubes, and observations on pre-boiler corrosion and its general effect on the terminal temperatures in the feedwater heaters.

In general, boiler water conditions were controlled without serious or prolonged periods out-of-specifications. Some difficulties were encountered during periods where the plant experienced transient power or temperature conditions. This was particularly true of the 1A boiler.

Primary-to-secondary leakage continued to exist in the 1B and 1C boiler (Reference 1) during Seed 2. Leakage was detected in the 1A boiler in November, 1960, and continued throughout Seed 2 life. The total leakage of all boilers did not create operational problems; detectable quantities of radioactivity in the secondary systems (excluding boilers) could only be measured after extensive concentration of samples.

Refueling - Layup

For the extended outage of approximately five months required by refueling, the layup of the boiler and condensate system differed from that used for short outages. For this extended layup, a boiler was drained at 200°F, the heads of the steam drum were removed, and the entire unit was dryed by fans. After an inspection, the boiler was resealed and purged with nitrogen gas (to minimize corrosion) until the oxygen content in the boiler was less than one percent by volume. The nitrogen was introduced through the main steam lead and vented out of the blowdown line. For the remainder of the layup period 1-1/2 psig of nitrogen pressure was maintained on the boilers. The oxygen concentration was maintained at less than 1 percent by volume without difficulty.

The condensate boiler feed, extraction steam systems were drained under a nitrogen blanket and maintained at 1-1/2 psig pressure during the entire layup. The system contained less than 1 percent oxygen during the entire period.

Steam Generator Performance

Control of boiler water chemical conditions were easily maintained during Seed 2 except during station startup following refueling and subsequent startups of 1A boiler. Following refueling, approximately one week was required for boiler water treatment stabilization during which time frequent chemical additions were needed. The abnormal chemical additions were attributed to abnormally high concentration of corrosion products in the boiler feedwater. A typical analysis of iron and copper during a station startup (6-20-61) can be seen in Table V-I. A 20-fold increase in iron is noted and a factor of 100 increase in copper. This pre-boiler corrosion collects in the boiler where it forms ferric phosphate and magnetic iron deposits. The formation of the very insoluble ferric or other possible metallic phosphate probably accounts for much of the high rate of loss of phosphate.

TABLE V-I

CONDENSATE SYSTEM CORROSION

	Hotwell		Boiler Feed	
	Fe	Cu	Fe	Cu
12-16-60	0.11 ppm	0.017 ppm	0.24 ppm	0.021 ppm
12-20-60	0.06 ppm	0.015 ppm	0.076 ppm	0.013 ppm
12-22-60	0.054 ppm	0.014 ppm	0.098 ppm	0.023 ppm
12-27-60	0.054 ppm	0.091 ppm	0.091 ppm	0.021 ppm
12-30-60	0.19 ppm	0.021 ppm	0.066 ppm	0.021 ppm
1-31-61	0.087 ppm	0.023 ppm	0.082 ppm	0.018 ppm
1-8-61	0.11 ppm	0.023 ppm	0.012 ppm	0.027 ppm
6-20-61	2.67* ppm	2.40* ppm		

* Sample taken during station startup after 6 days of cold layup.

Hideout of chemicals in the 1A boiler was not noticed until March, 1961, near the end of Seed 2 life with three loops in service. During the early part of the month, the 1A boiler required more than the usual amount of chemicals and there was a perceptible increase in pH. Later, upon shutdown of the generator, the boiler water concentration increased about 25 percent. If there were hideout prior to this time, it was milder and probably masked by the usual changes in boiler water composition incidental to continuous sampling; the chemical additions were not sufficiently different to arouse suspicion. The cause of this hideout condition has not been established at this time; further investigations are continuing.

Hideout in 1A Boiler

March 30, 1961

Sample Time	Part per Million Sulfite	Part per Million Phosphate	mmhos Conductivity	pH at 25°C
0001	82	220	1220	10.85
1600	57	194	1120	10.90
1616	58	195	1133	10.90
2000	42	253	1070	10.71
2345	45	258	1055	10.85

* Station off line at 1603

Typical operating data for the 1A and 1B boilers are shown in Figure V-22. This condition in the 1A boiler was not of sufficient magnitude to create a control problem when at steady-state power.

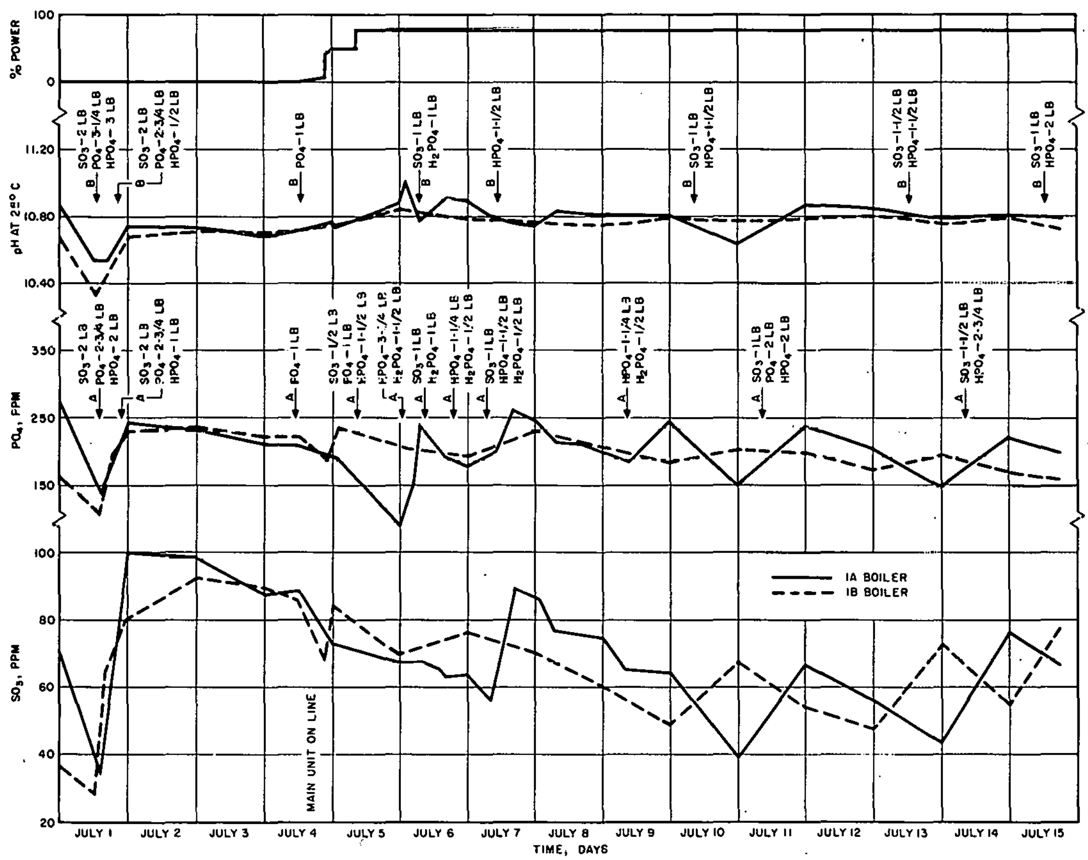


Figure V-22. 1A and 1B Boiler Water Conditions (July 1 to 15, 1960).

Primary to Secondary Leakage

During Seed 1 operation, the incidence of primary-to-secondary leakage was established in the 1B and 1C boilers. The leakage of these boilers was confirmed during Seed 2 operation.

On November 6, 1960, during station start-up, primary-to-secondary leakage was detected in the 1A boiler by the presence of I^{133} (20.8 hr half-life) in the boiler water. To establish the leak rate, the reactor coolant purification system was removed from service to allow the I^{133} to equilibrate at a higher level of activity and, hence, increase the sensitivity of the leak rate estimation. The purification system remained isolated from November 16, to November 25, 1960; based on I^{133} concentration in the reactor coolant and in the boiler the following average leak rates were calculated for the boilers in service:

1A	1B	1C
0.61 ml/min	0.02 ml/min	0.53 ml/min

During the remaining 9 months of the Seed 2 operating period, the leak rate of the 1B and 1C boilers decreased to less than detectable amount while the 1A boiler leakage increased. The last measurement indicating a leak rate of 2.6 ml/min. $\pm$10 percent in the 1A boiler

No leakage was detected in the 1D boiler to the end of Seed 2 operation. On occasion, Iodine-131 (8.08 day half-life) has been detected in the 1D boiler; its source was believed to be carryover from the other boilers since it is possible to detect I^{131} in the 1A boiler steam by using an ion exchange technique to concentrate the activity. Since the steam from all boilers flows to a common turbine and is returned as water via a common feed system, carryover in one boiler will affect all.

Although there has been detectable leakage in three of the four boilers during Seed 2 operation, this leakage has not created problems in the turbine plant. Radiation levels have remained at normal background values; smear samples from components have not shown the presence of radioactive contamination.

Pre-Boiler System Corrosion

Shippingport is a saturated steam turbine station. The iron and copper pickup in the condensate and boiler feed system have been consistently higher than those experienced at conventional stations in the Duquesne Light Company System where morpholine and hydrazine have been used to reduce pre-boiler corrosion. The use of corrosion inhibitors for the condensate system has not been authorized since these inhibitors could possibly have a detrimental effect on the reactor if leakage to the reactor coolant system occurred.

There is increasing evidence that the high level of iron in the condensate is contributing to the rapid fouling of the tubes of the feedwater heaters and to the accumulation of iron oxide sludge along the bottom of the boiler heat exchangers. The terminal temperature difference (TTD) of the heaters are plotted in Figures V-23 and V-24. The TTD of comparable heaters at the Duquesne Light Company's Power Station are much lower than Shippingport even after 8 years service and have been essentially constant during the last 5 years of trial and adoption of preboiler treatment. At the

Duquesne Light Company's Phillips Power Station, TTD of comparable heaters increased to 40°F without treatment whereas with treatment the TTD is only 18°F after 3-1/2 years operation. The iron and copper content of the condensate and feedwater at Shippingport are summarized in Table V-I.

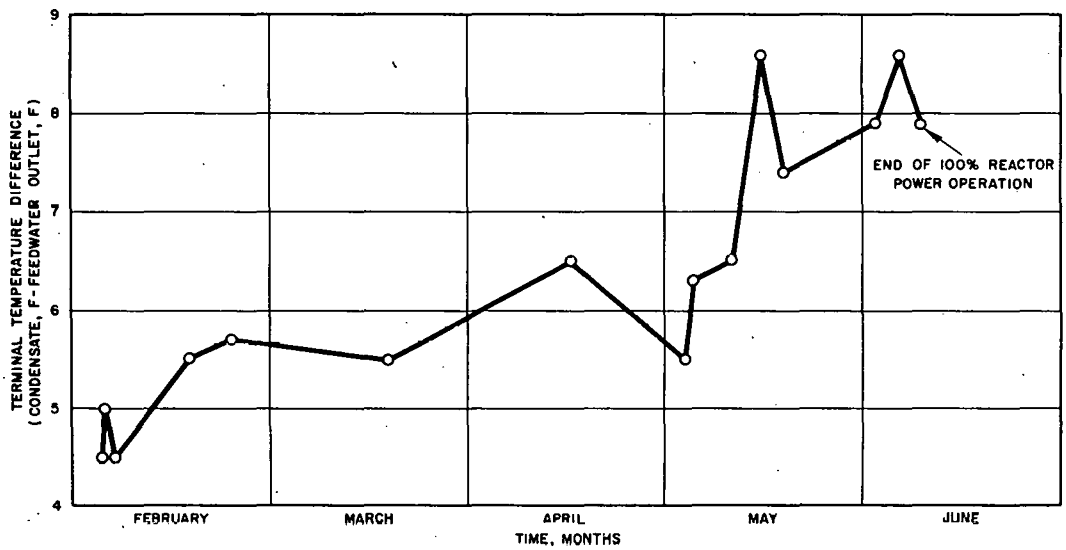


Figure V-23. Terminal Temperature Difference of 1A FW Heater after Retubing.

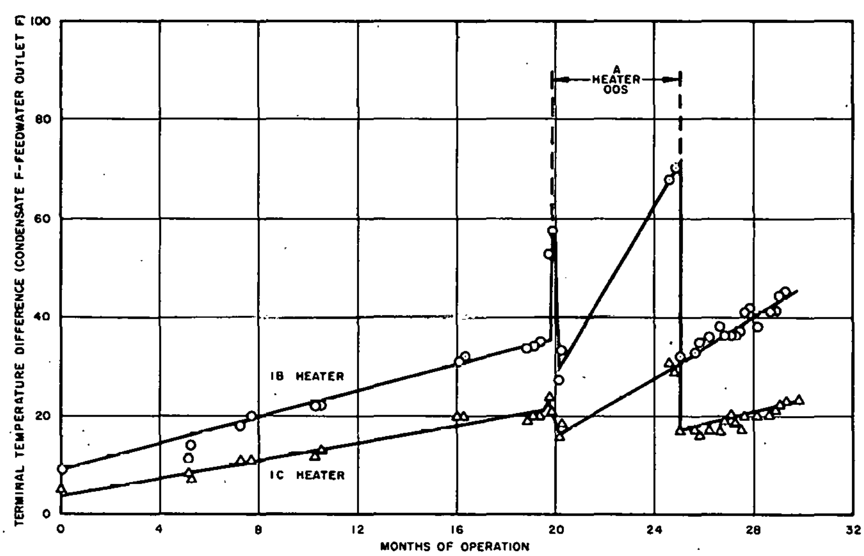


Figure V-24. Terminal Temperature Difference of Feedwater Heaters -- No. 1 Main Unit.

The accumulation of iron oxide sludge in the bottom of the boiler heat exchangers has been recognized as a problem. At attempt has been made to remove sludge accumulations by bi-weekly, fast blowdowns of the boilers. The maximum blowdown rate attainable has been 5000 lb/hr at operating pressure and, judging by the turbidity of the effluent, has not been of benefit in moving this sludge. The accumulation of sludge has been suspected as a possible cause of chemical hideout noted in 1A boiler during steady-state full power operation.

At the conclusion of Seed 2 operation, the steam drums of the four main unit heat exchangers were opened for the annual Commonwealth of Pennsylvania pressure vessel inspection during September. The inspection of the steam drums revealed that the largest quantity of deposit, consisting of uniform black magnetic material, was contained in the 1B and 1C steam drums. This deposit was not considered unusual or excessive for operating conditions. The deposit contained in the 1A and 1D steam drums was somewhat greater than the quantity noted in the previous annual inspection. The 1D steam drum showed evidence of atmospheric corrosion above the water level due to long periods of cold, wet lay-up; however, there was no evidence of pitting.

Results of an inspection of the shell side of the 1A heat exchanger showed that an accumulation of grannular deposit was present. (About 1180 grams of deposit was collected when the bottom inlet handhole was removed.) This deposit was somewhat different in nature from the material found in the steam drum. The inspection port was located at approximately 5 o'clock looking in from the inlet end. A flexible 12-inch probe was inserted into this handhole in an effort to determine if there was an impaction of sludge against the inlet tube sheet, as had been noted in the previous annual inspection. There was no evidence of such an impaction. The scaly nature of the deposit collected indicates that the deposit had been baked onto the lower tubes; some of this deposit was still evident on the lower tubes. The deposit on the 1A heat exchanger tubes was tightly adherent.

The tubes on the shell side of the 1C heat exchanger were covered with a large amount of loosely adherent deposit very similar in nature to that found in both the 1A and 1C steam drums. Since the inspection port used for this inspection was located at approximately 4 o'clock looking in at the inlet and, no measurement could be made regarding the quantity of deposit in the bottom of the heat exchanger.

Air Ejector Tube Leakage

In June, 1961, with the station operating at full power, leakage was detected in the after-condenser section of the air ejector. The volume of leakage became so great that it exceeded the capacity of the after-condenser section drain, which discharges to the drip tank. The drain line was cut and the drains were diverted to a floor sump on the lower level. Here the water (being at about 160°F) released considerable vapor, which had the characteristic smell of hydrogen sulfide. A qualitative test with lead acetate paper confirmed the presence of hydrogen sulfide in the atmosphere. Samples of the drain water had pH values of 5.3-5.5, considerably below that of pH 6.5-7.5 of normal condensate. The sulfide measurements indicated the concentration was below 0.05 ppm (the lower limit of detection).

At the next station outage (4 days later) the air ejector was opened. Four leaking tubes were located; several leaking and non-leaking tubes were removed. The badly externally corroded condition of these tubes led to a decision to replace all 76 tubes in the after-condenser section. At the same time, sampling lines were installed on the drain line and the air ejector vent line. These had heretofore been piped solid, the one to the drip tank and the latter to the container air ventilation discharge stack.

These tubes were manufactured from arsenical copper (3/4 OD x 0.049 inch wall). Inspection disclosed that all the removed tubes had experienced external corrosion, more severe at the cooling water (condensate) inlet end. The corrosion was apparently due to one or more of soluble or slightly ionizable gases such as O_2, NH_3 and H_2S, which are known to be present. The tightly adherent deposit removed from the tubes contained over 3 percent sulfur as sulfide. Oxygen and carbon dioxide are from the atmosphere; ammonia is the decomposition product of the nitrogenous organic compounds that leak through the make-up demineralizer. The ammonia of the condensate is distilled in the boiler and shows up as a gas concentrate in the air ejector vents and drains. A concentration factor of 10 to 1 to 30 to 1 is not unusual for ammonia under these conditions.

Hydrogen sulfide, which was detected at the drain, would behave and concentrate similar to ammonia. The source of H_2S is presumable the auto-oxidation of the sulfite in the boiler water. Hideout conditions are known to be favorable to the breakdown of sulfites. Sulfites are ordinarily not thought to break down appreciably at temperatures below 535°F (900 psi saturated); however, it is possible that if there are stagnant conditions in the sludge covering heat exchanger tubes, some sulfite decomposition could occur.

The air ejector was retubed with admiralty alloy tubes, which should be markedly more resistant to sulfide attack. A design change was made to the air ejector by relocating the vent line, which will improve the removal of these "gases" and minimize their concentration in the drains.

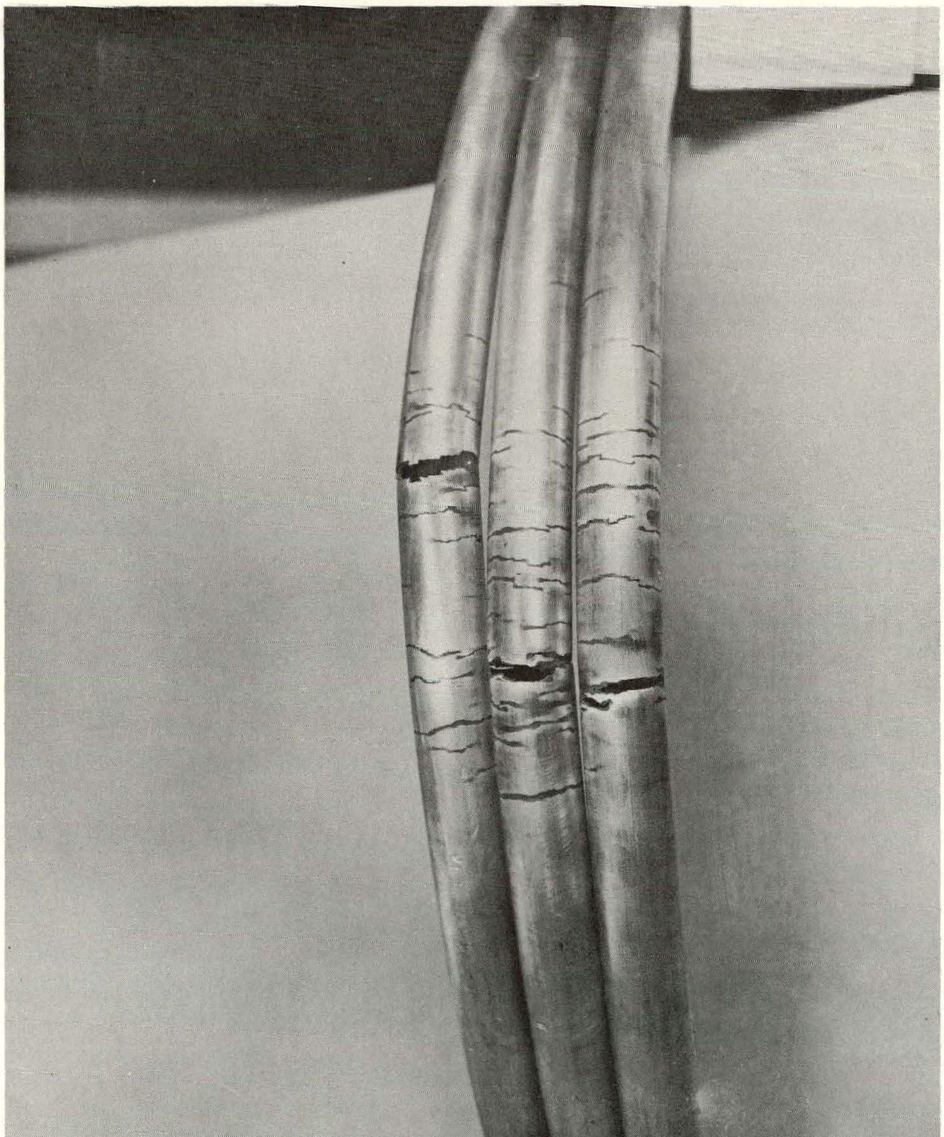


Figure V-25 Circumferential Cracking -- 1A Feedwater Heater.

After the plant was returned to service and for the ensuing period of reduced capacity operation occasioned by seed burnout, the pH of the drains was the same as that of the condensate, and no hydrogen sulfide could be detected either in the drains or in the air ejector vent. Presumable, the boiler condition that engendered the sulfite decomposition were not encountered during the reduced power runs.

Feedwater Heater Leakage

The cracking and leakage of the tubes in the feedwater heaters required considerable heater outage for leak testing and retubing. Leakage was first detected and confirmed in the 1A heater early in 1960. By July, 1960, 91 leaking tubes were plugged and the heater was removed from service for retubing. By April, 1961, the 1B heater contained 53 leaking and plugged tubes; consequently, it was scheduled for retubing during the Seed 2 refueling. All of the tubes in these heaters are arsenical admiralty alloy.

Examples of the cracking are shown in Figure V-25. Three different locations and forms of cracking have been identified.

1. Cracks originating on the outside tube surface near the first tube sheet from the U-bend. Only a few tubes have experienced this type of failure.

2. Cracks originating on the outside tube surface in the U-bend portion of the tubes, and having characteristics of stress corrosion and corrosion fatigue failure.

3. Cracks originating on the inside surface of the tubes and having characteristics of stress corrosion or corrosion fatigue or, perhaps, fatigue alone.

The variability in cause of cracking results from the variety of cracks. Some were broad and blunt, some were fine and both intergranular and transgranular. The metal adjacent to some inside cracks appeared to be necked down implying cyclic high-tensile stresses. Some tubes had a multiplicity of failures indicating simultaneous progression of cracks through the tube wall. The ammonia content of the condensate has not been above 0.2 ppm; however, some concentration undoubtedly occurs in the vicinity of the heater vents. The cascading heater drains give rise to thermal stresses, which were calculated to be as high as 9600 psi. These stresses superimposed on the operating stresses, resulted in local stresses near the yield point of the material. These several factors can be used in various combinations to account for the tube failures.

Aside from retubing, the only remedial action was the increase in the venting of the heaters and modification of operating procedures to avoid thermal shock of the tubes. This requires pumping of the 1B and 1C heater drains to the boiler feed system during certain phases of operation.

REFERENCES

1. DLCS 364, "Shippingport Operation Start-up to the First Refueling".

2. WAPD-BT-12, "One Year Operating Experience at Shippingport".

CHAPTER 5

RADIOACTIVE WASTE DISPOSAL SYSTEM

Introduction

The operating experiences for the first 22 months of the radioactive waste disposal plant as Shippingport were detailed in an earlier report (Reference 1). This report includes the operating experience during Seed 1 - Seed 2 refueling in addition to Seed 2 operation. The volumes and activities of the various classes of wastes are summarized graphically (Figures V-26-29) and compared with the design criteria. On occasions the design volumes and activities exceeded design but were readily processed utilizing the capabilities of the entire RWD System. Although there have been specific equipment problems associated with this system, it has never limited power production nor has the State regulated limits governing the discharge of activity to the environment been exceeded.

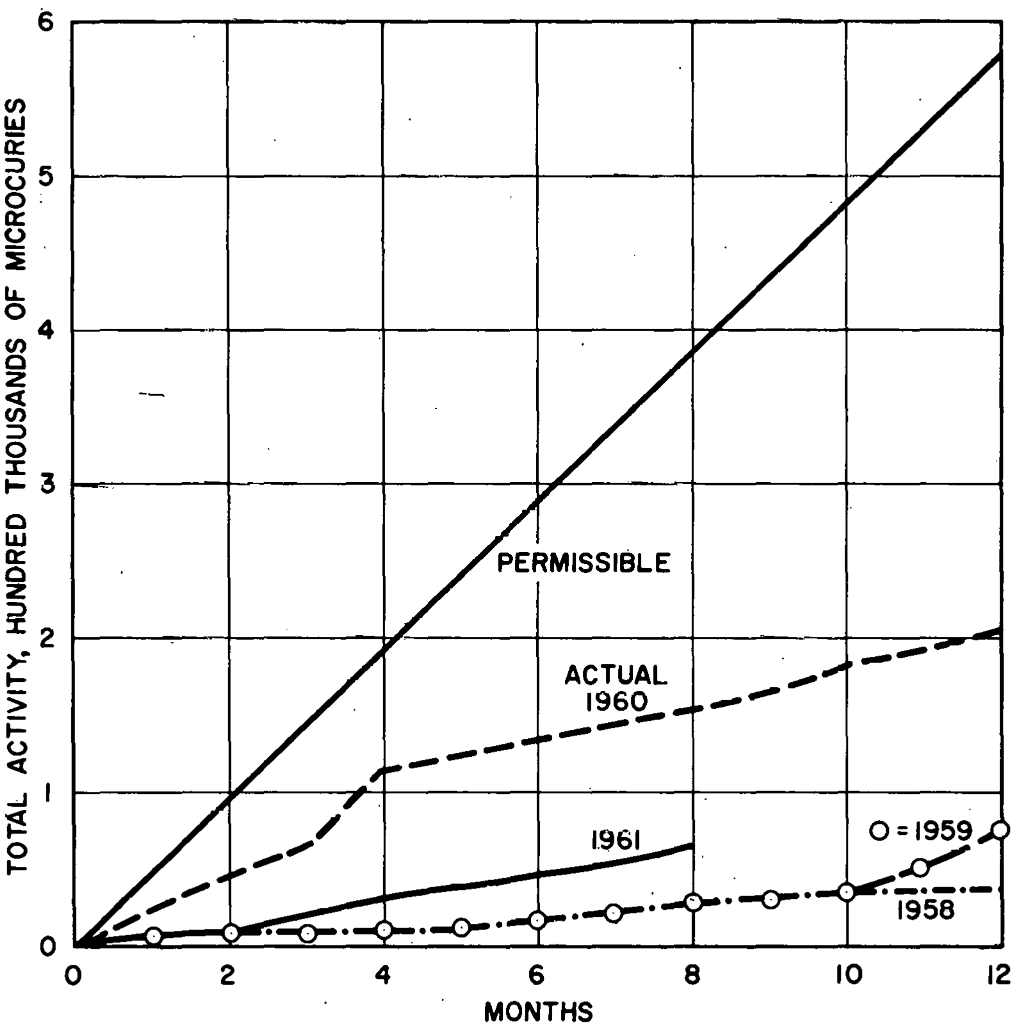


Figure V-26. Comparison of Actual Discharges of Radioactivity to the Ohio River during 1958 to 1961 with Average Permissible Discharges Allowed by Permit No. 1832 (cumulative totals by months).

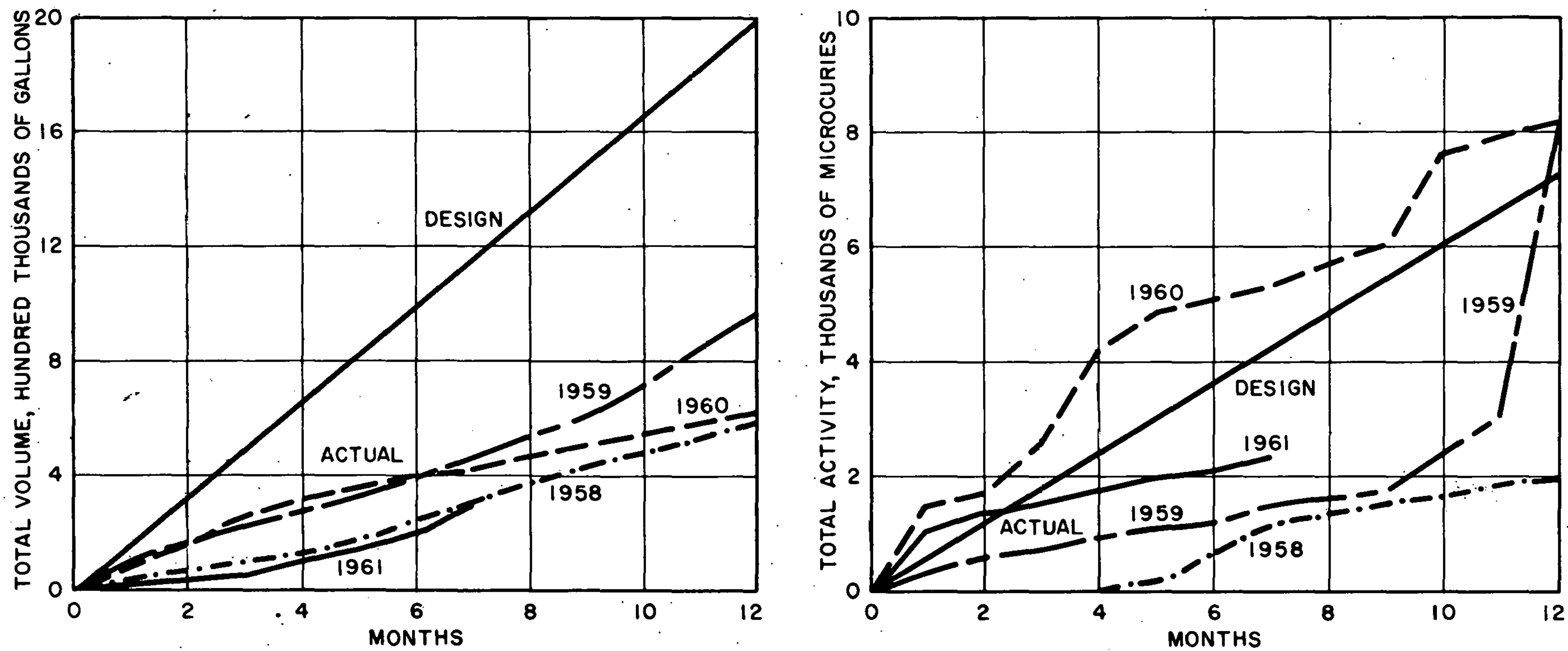


Figure V-27. Comparison of Actual and Design Cumulative Totals of Volume and Activity of Nonactive (A) Wastes Discharged to the Ohio River from 1958 to 1961.

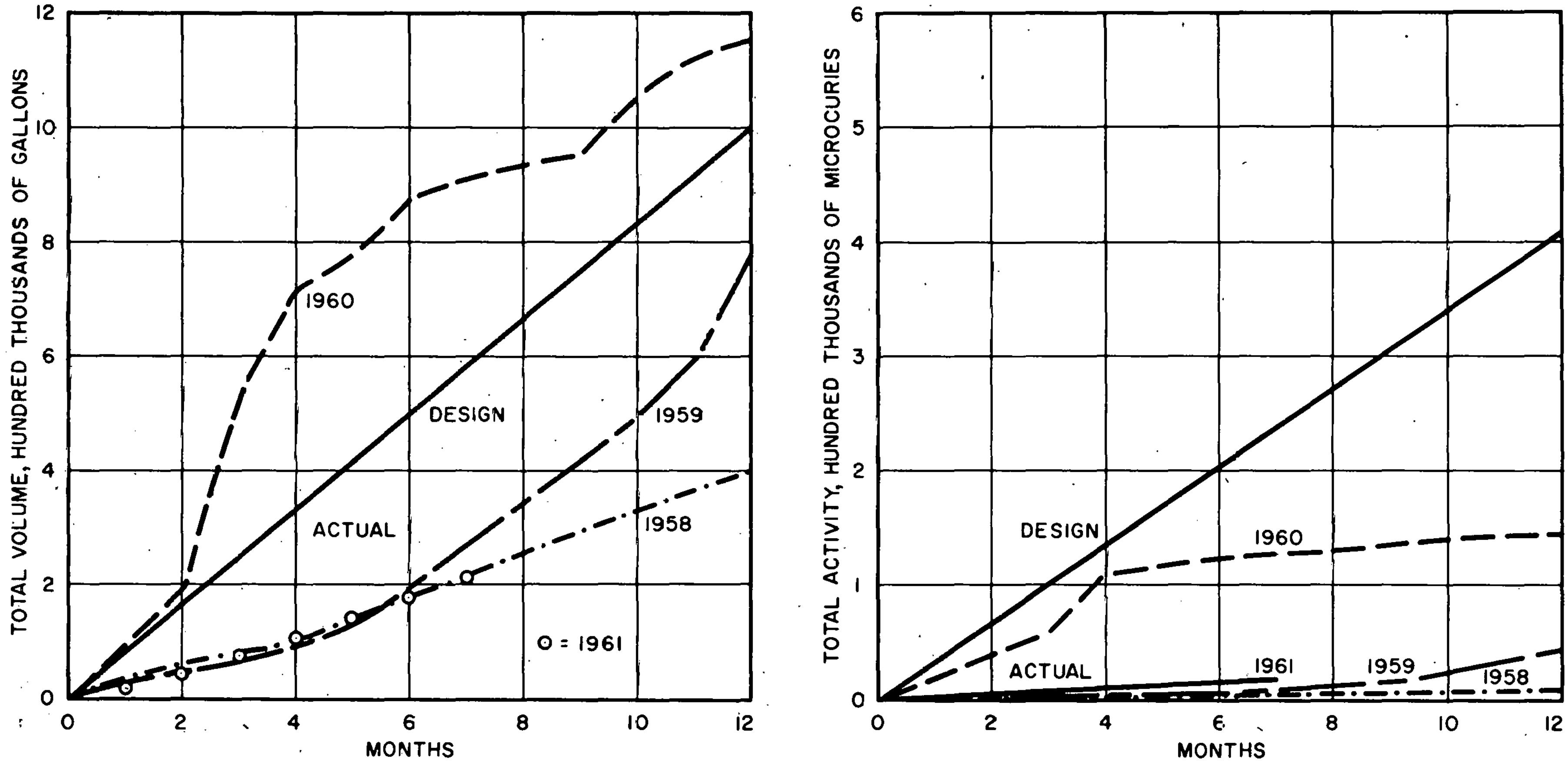


Figure V-28. Comparison of Actual and Design Cumulative Totals of Volume and Activity of Special Monitored (B) Wastes Discharged to the Ohio River from 1958 to 1961.

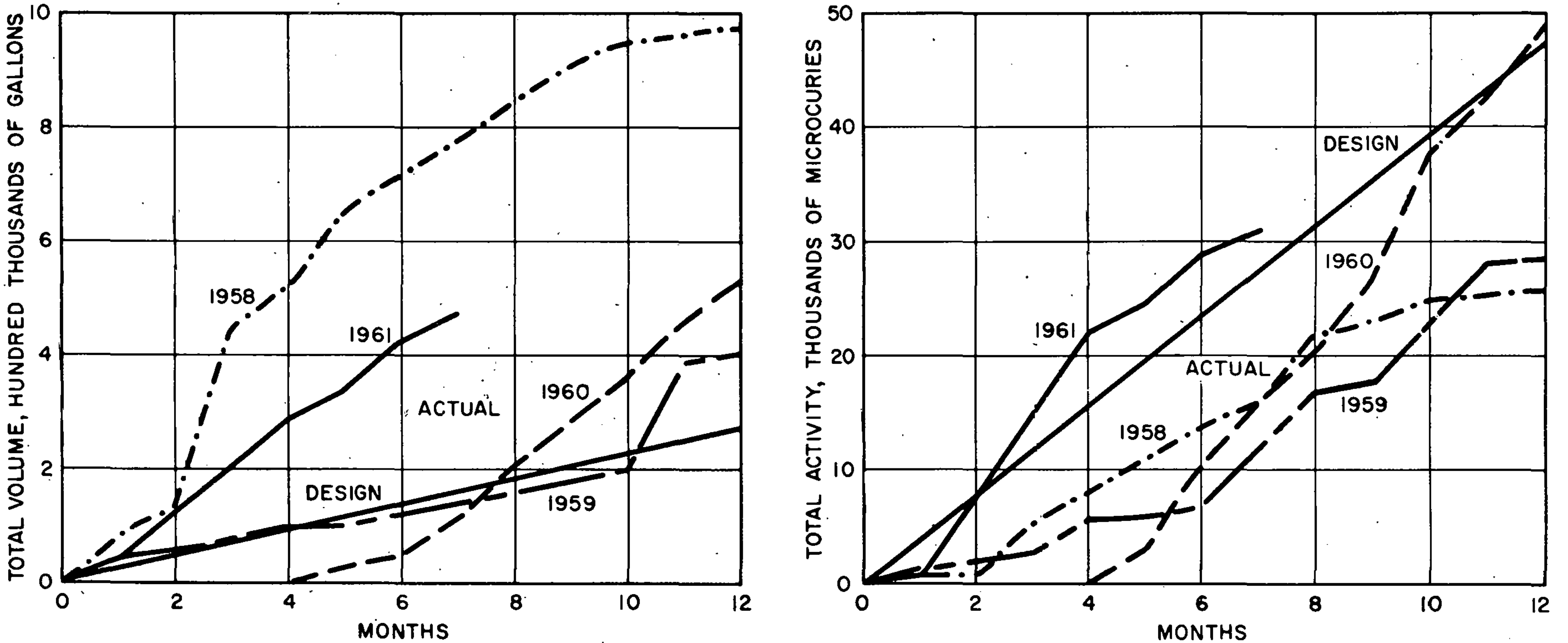


Figure V-29. Comparison of Actual and Design Cumulative Totals of Volume and Activity, Excluding Tritium, for Reactor Plant Effluent Discharged to the Ohio River from 1958 to 1961.

Design Criteria and Design Assumption

The original design criteria and design assumptions for an operating reactor plant are discussed in detail in Reference 1. The plant criteria for discharging activity to the environment are within the new radiation and contamination limits as promulated by the International Council on Radiation Protection, National Council on Radiation Protection, National Bureau of Standards Handbook 69 and those of the Federal Radiation Council as expressed by the Radiation Protection Guide and Radioactive Concentration Guide.

Nonactive Liquid Wastes from Reactor Service Building

The nonactive liquid waste disposal system contains and processes all liquid wastes from the showers, nonactive laboratory sink drains and the nonactive laundry. There have been no problems associated either with the volume or activity of this waste during Seed 2 operation. During Seed 1 - Seed 2 refueling there were a few occasions when transfer of this waste to a special waste tank or chemical waste tank was necessary because of activity. An investigation pointed out that the laundering of outer garmets caused the high activities in the nonactive wastes. When the outer garmets were laundered separately, and the wastes discharged to the special waste tanks, the problem of "hot" nonactive wastes was eliminated.

The data for this system are presented in Figure V-27. The volume of wastes processed in this system has been essentially constant from year to year and well below design estimates. From the standpoint of activity, the significant point is the large increase which became evident in November 1959 and continued through April 1960. This "abnormal" condition was due entirely to the Seed 1 - Seed 2 refueling operation.

Special Liquid Waste from Reactor Service Building.

The special liquid waste disposal system contains and processes all liquid wastes from the laundry and special laboratory drains in the reactor plant service building. These wastes are normally discharged at a controlled rate to the Ohio River or to the chemical waste tanks if the activity is too high. There have been no problems associated either with the volume or activity of this type of waste during Seed 2 operation. However, the demands placed on this system in terms of both volume and activity during the Seed 1 - 2 refueling period and subsequent cleanup and decontamination operations has, on occasion, exceeded the instantaneous capability of the special waste system. This is evident in Figure V-28 by the slop of both curves for the refueling period November 1959 to April 1960. Because of its limited design capacity of 2400 gallons per day, the evaporator could not serve as a backup for processing special wastes which could exceed limits for allowable discharge to the river, especially when the accumulation rate of these wastes is 3500-7000 gallons per day. Under such abnormal conditions the waste has been diverted to a surge tank for temporary storage. This method of handling produced a subsequent problem. Normally surge tanks handle water with relatively low concentrations of ionized salts and are processed by ion exchange. The dome shaped bottom of the tank does not permit complete withdrawal of contained liquid; therefore, residual detergent laden special waste contaminates the reactor plant effluents necessitating excessive use of ion exchange resin during future processing.

Reactor Plant Effluents

The reactor plant liquid waste system collects, processes, and disposes the various wastes from the reactor coolant system. Storage capacity is provided in the form of four surge tanks having a combined storage capacity of 116,000 gallons. However, 30,000 gallons free storage space must be kept available for operation of the safety injection system. This limitation reduces the net storage capacity to 86,000 gallons. The system design was based on filling a tank in an average of 45 days, decay for 45 days and discharge in 7 days. Data for these wastes are presented in Figure V-29. Experience has shown that the activity of plant discharges is lower than the design assumption, but the volume has been greater by a factor of 3. This unexpected volume is also reflected in the cumulative activity being near design. The basic reason for the abnormal volumes is directly related to thermal cycling of the plant and to relief valve and hydrostatic testing. A surge tank is filled in approximately 12 days and is processed by ion exchange and gas stripping to hold up tanks for first testing in approximately nine days. It should be pointed out that the gas stripper has been effective for removal of radioactive gases and that the 45-day decay time is not necessary at this time.

Plugging of the inlet filter of the ion exchangers continues to be a problem; backflushing is used to restore operating flow rates but the effect of this backflush on bed capacity is not known.

Chemical Wastes

The chemical waste system is designed to handle liquid wastes having an activity too high for discharge to the river or a dissolved solids content too high to be economically processed by ion exchanger. The system consists of two receiving-mixer-neutralizing tanks and a vapor compression evaporator. Because chromated water from other sources must be diverted to the chemical wastes, the volume of these wastes was higher than the original design assumption of one tank every 15 days. During Seed 1 operation, the chromated wastes were generally low in activity, thus permitting chemical precipitation of chromate and discharge of the decant fluid. However, the activities of various wastes have increased during Seed 2 to a level which precludes this operation; evaporation is mandatory.

The evaporator is rated at 2400 gallons per day, if it is operated continuously. Reliable continuous operation has not been attained despite consultation with the manufacturer, modifications to the system, excessive maintenance, and the use of extra care and manpower during operation. The evaporator experienced repeated feed line clogging, which is corrected by backflushing to the feed tank, thereby increasing the volume of waste without disposing of the material that caused line plugging. This is especially prevalent during concentrate evaporation of the chromate precipitate slurry or the lint-laden laundry water of special wastes.

Laundry water and decontamination water contain detergents and chelating agents, so that the foaming problem is always present. Anti-foam agents have been moderately effective, but the dosage must be determined experimentally for each batch. The anti-foam agents suppress excessive foaming to the point where carryover is prevented; however, it is necessary that an operator remain at the evaporator to observe the process through the sightglasses and relay any corrective action to the control board. Level-control instrumentation does not recognize the foam and hence does not transmit a signal to the feed regulating valve to reduce feed. Upon observation of foam, corrective action consists of lowering the evaporator level control point, resulting in reduced feed and adding additional anti-foam agent held on standby in the acid cleaning tank. This generally suppresses the foam for a time, but there is no way to predict the initial appearance of foam or its recurrence.

Removal of evaporator bottoms is often difficult, particularly when the ambient air temperature is cold. Removal can be minimized by purging out the concentrate at intervals of about 16 hours, (before the 30 per cent solids point is reached), thereby increasing the number of drums destined for land burial.

As mentioned under special wastes, the evaporator is not sized to be considered a backup for the processing of these wastes. The production of special wastes above discharge limits invariably is associated with some operation that is of several days or weeks duration so that a 3500 gallon per day production rate would not be unusual, whereas there have been months when chemical wastes were produced at rates that would require 50 per cent evaporator operating time at full load. Although there have been evaporator difficulties throughout Seed 1 - Seed 2 refueling and Seed 2 operation, the overall waste disposal facilities have been able to handle all waste liquids.

Tritium

Lithium hydroxide is used in the reactor coolant system for pH control. Irradiation of $Lithium^6$ in natural lithium (7.4% by weight of Li^6) results in the formation of Tritium. Although Tritium is not normally a radiation hazard due to its short biological half-life (19 days) and its very low energy beta energy (18 Kev), it was considered desirable to minimize Tritium discharge to avoid any possible problems.

The use of $Lithium^7$ isotope, 99.997 per cent pure, both in the $Lithium^7$ hydroxide and $Lithium^7$ regenerated cation resin in the mixed bed ion exchangers of the purification system has materially reduced the tritium discharges from the plant. The tritium concentration in the reactor coolant is now 2-3 μc/liter which is near the lower limit of radio-chemical detection. Prior to the use of $Lithium^7$, the maximum tritium concentration in the coolant was approximately 300 μc/liter. Because reactor coolant is diluted in the surge tanks by water from other sources, the reported tritium concentration for the contents of surge tanks is generally one-half as great as that found in the reactor coolant samples.

The following are the total yearly tritium discharges from this plant to the Ohio River:

1958 - 50 curies
1959 - 64 curies
1960 - 99 curies
1961 - 12 curies (7 months)

For the last four of the seven months in 1961 when Lithium-7 was used, the total tritium discharged was less than 2 curies.

Gas Stripper

During the life of Core 1 Seed 2, the gas activity in the water being processed reached a level sufficient to warrant the use of the gas stripper. This greatly increased the processing time as the stripper is capable of processing only 3 gpm, in contrast to 10 gpm when ion exchangers are used alone. No unusual operating or maintenance has arisen in connection with the gas stripper.

Hydrogen Burner

The original catalyst provided in the hydrogen burner became inactive within a short period of time, varying from 3 to 10 operating hours. The catalyst then had to be removed and thoroughly cleaned or replaced. A different type catalyst (activated palladium on aluminum oxide carrier) was installed in parallel with the original burner and, to date, after approximately 12 months of operation no replacement or cleaning has been necessary. When the catalyst becomes inactive it is necessary only to purge the burner with nitrogen for a few minutes and the unit is back in service. The original catalyst is used only for standby purposes.

Incinerator

The earlier problems with the incinerator continue. The incinerator was removed from service because of excessive smoke discharging into the atmosphere and into the radioactive waste disposal building. The scrubber is ineffective because of the inability to maintain a constant water level. Modifications were made to the scrubber but tests have not been run to determine the effect of these modifications. At the end of Seed 2 lifetime, the incinerator was out of service pending resolution of all problems. Contaminated combustible wastes are placed into 55-gallon fiber drums and shipped for land burial. Approximately 4 to 5 drums of contaminated combustible wastes are accumulated per week.

REFERENCES

1. "Shippingport Operations; From Start-up to First Refueling December 1957 to October 1959," DLCS-364.