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ANNULAR CORE PULSE REACTOR UPGRADE QUARTERLY REPORT
APRIL - JUNE 1977

MASTER

Reactor Development and Applications Department

Prepared by Sandia Laboratories, Albuquerque,
New Mexico 87115 and Livermore, California 94500
for the United States Nuclear Regulatory Commission
under ERDA Contract AT(29-1)-789.

Printed September 1977

Sandia Laboratories
Nuclear Fuel Cycle Programs

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ANNULAR CORE PULSE REACTOR UPGRADE. QUARTERLY REPORT*
April - June 1977

Submitted by
Reactor Development and Applications Department
Sandia Laboratories, Albuquerque, New Mexico 87115

Approved:
Manager, Reactor Development and Applications Department

Director, Nuclear Fuel Cycle Programs

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ANNULAR CORE PULSE REACTOR UPGRADE QUARTERLY REPORT

INTRODUCTION

During FY75, Sandia Laboratories conducted an Experimental Fast Reactor Safety Research Program funded by the Nuclear Regulatory Commission/Division of Reactor Safety Research. A portion of this program involved a feasibility study for performance improvement of the Annular Core Pulse Reactor (ACPR Upgrade). The progress on the ACPR Upgrade during FY75 is described in References 1, 2, and 3. Beginning with FY76, the ACPR Upgrade progress is reported in quarterly reports separate from reports on the Experimental Fast Reactor Safety Research Program.[4-10]

Funding for the ACPR Upgrade is being provided jointly by NRC/DRSR and ERDA/DMA since the improved reactor will be beneficial to the programs of both agencies.

The object of the ACPR Upgrade is to arrive at a reactor modification which will provide an increased pulsed neutron fluence in the irradiation cavity without an excessive degradation of the pulse duration. The upgraded reactor will also have an increased steady-state neutron flux. The initial approach to the upgrade modification involved a two-region core concept. The inner region, surrounding the irradiation cavity, consisted of a high-heat-capacity fuel capable of sustaining a large fission energy deposition. The outer region consisted of a uranium-zirconium hydride fuel similar to the present ACPR fuel. During a previous quarter, the high-heat-capacity fuel was chosen as $BeO-UO_2$ and in-pile tests were conducted with $U-ZrH_{1.5}$ samples. Posttest examination of the hydride samples indicated that further fuel development would be required for the U-ZrH fuel. Since such development was not feasible due to schedule and budget considerations, calculations were performed for a core consisting entirely of $BeO-UO_2$ fuel elements. These results indicated that an all $BeO-UO_2$ core would produce a greater improvement for the ACPR Upgrade than a two-region, $BeO-UO_2$ and ZrH core. Hence, the decision was made to utilize a single-region core of $BeO-UO_2$ fuel elements. The reactor modifications will utilize the majority of the existing structure and will be accomplished in a relatively short time.

The ACPR Upgrade project was originally divided into nine tasks to improve management of the overall project and to maintain close control of the project budget. As the project completed or eliminated a task, it was dropped or combined with another task. This report discusses the progress on each task in a separate chapter. The individual tasks and a brief description of each are given below.

Task 1. Safety, Documentation, and Compliance (J. A. Reuscher, Supervisor)

This task involves the preparation of the safety analysis report and the technical specifications for the upgraded reactor. These documents must be submitted to ERDA Division of Operational and Environmental Safety for review and approval prior to startup of the reactor. Compliance with the requirements contained in 10CFR50 is a part of this task; these include an independent design review and quality assurance program. In addition, the initial test planning for the reactor is a part of this task.

Task 2. Core Nuclear Design (J. A. Reuscher, Supervisor)

This task includes core neutron physics studies, determination of control rod configurations, and the prediction of experimental conditions. Correlation of calculational techniques with the present ACPR is included.

Task 3. Console Development (J. E. Powell, Supervisor)

This task is concerned with the design, development, and procurement of a control system which follows IEEE 279 standards.

Task 4. Mechanical Design (J. H. Davis, Supervisor)

Mechanical design activities for the project include the cooling system, the increased hoist height, the cavity ventilation system, drive mechanisms for the control and transient rods, and the control rod design.

Task 5. Fuel Element Design (J. A. Reuscher, Supervisor)

This task interfaces with the fuel material fabrication task (Task 6) and includes the stress analysis and heat transfer studies for design of the $BeO-UO_2$ fuel elements. The fuel element demonstration tests are also a part of this task. Procurement of the clad, niobium liner, and insulators is part of this task.

Task 6. $BeO-UO_2$ Fuel Element Fabrication (C. H. Karnes, Supervisor)

This task was originally the Primary Fuel Material Studies and involved the development studies required to prooftest the $BeO-UO_2$ fuel material. Since $BeO-UO_2$ was chosen as the fuel for the upgrade core, this task is concerned with production of the $BeO-UO_2$ fuel annuli and assembly of the fuel elements.

Task 7. Secondary Fuel Material Studies (C. H. Karnes, Supervisor)

The secondary fuel material was (UC-ZrC)-graphite which was to be used in the high-heat-capacity fuel element if the $BeO-UO_2$ did not prove feasible. This task involved the development of fabrication techniques, material compatibility studies, material property determinations, material analyses, and in-pile experiments. Since (UC-ZrC)-graphite was not chosen as a fuel, this task is no longer a part of the project.

Task 8. Driver Core Fuel Element (J. A. Reuscher, Supervisor)

The testing of the proposed outer core fuel material and the design of the driver core fuel element were the objectives of this task. This fuel was a uranium-zirconium hydride similar to the present ACPR fuel. Since the decision was made to use an all $BeO-UO_2$ core, this task was eliminated from the project.

Task 9. Diagnostic System (J. E. Powell, Supervisor)

This task involves the development of a fuel motion detection system for fissile experiments in the upgraded ACPR. Such a system allows the detection of molten fuel motion in a reactor experiment. Several schemes are under development and involve both in-core and out-of-core devices. Progress on this task is reported as part of Sandia's Fast Reactor Safety Research Program (Ref. 10).

Project Schedule

The schedule for the ACPR Upgrade is shown in Figure 1. This figure gives the major events in the project and projects an operational date (critical experiment) about March 1978. A detailed PERT analysis of the overall project has been conducted, and the critical paths have been identified. The PERT chart is too detailed to include in this report.

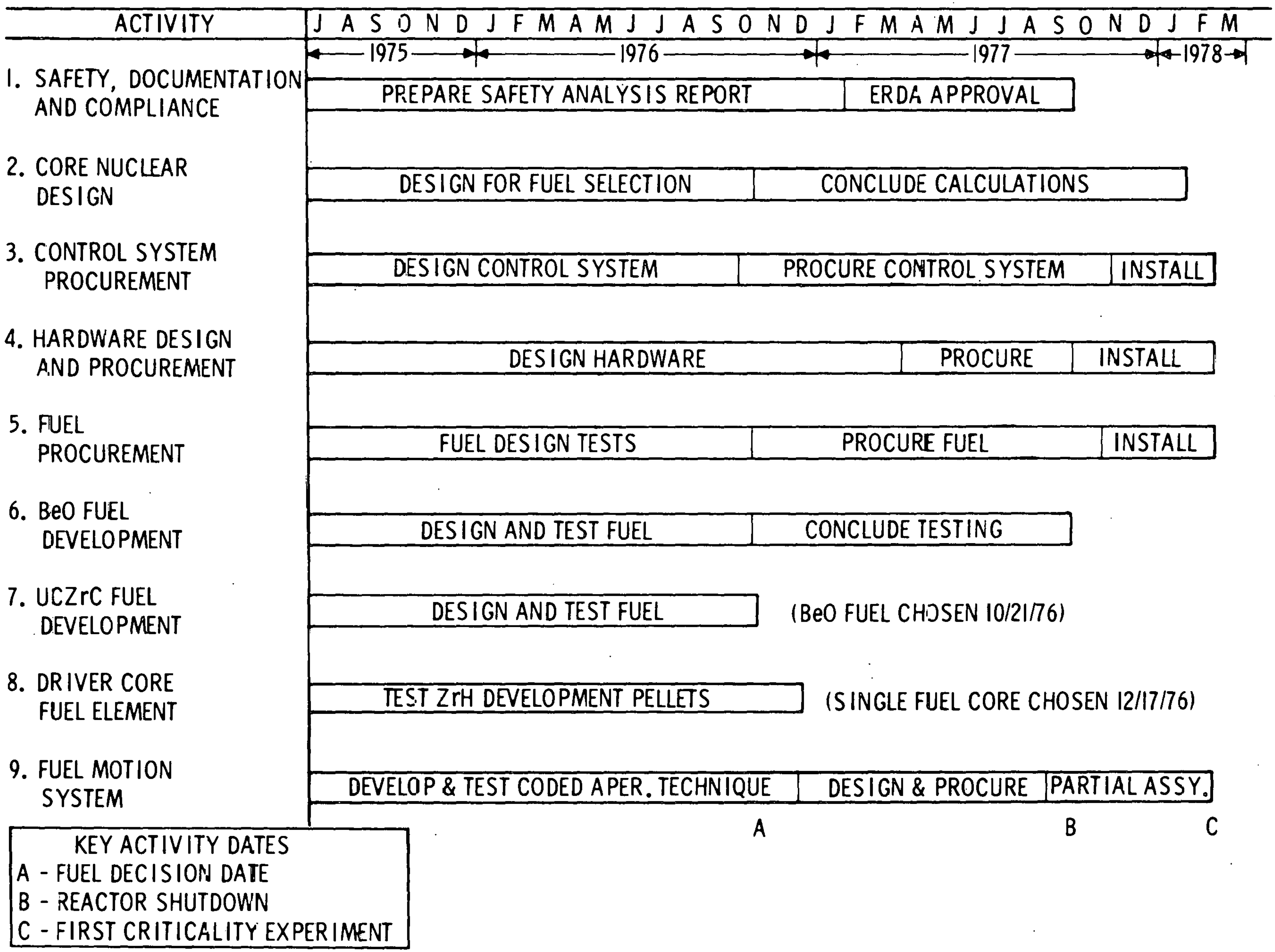

Figure 1. ACPR Upgrade Schedule

CHAPTER I

TASK 1. SAFETY, COMPLIANCE, AND DOCUMENTATION

J. A. Reuscher, 5450; A. C. Marshall, 5452

Introduction

This task involves the preparation of the safety documentation for the ACPR Upgrade, compliance with ERDA regulations concerning reactor design and construction, and planning of the initial tests and operations.

Safety Analysis Report (SAR)

The revised Safety Analysis Report for the ACPR Upgrade was submitted to ERDA in February 1977, and the extensive review and evaluation was begun during the second quarter of 1977. The review was conducted by ERDA Headquarters and Albuquerque Operations Office, by consultants at North Carolina State University, and by the Nuclear Regulatory Commission. By the end of June 1977 about ninety formal questions and comments had been received, and there did not appear to be any major safety issues raised by ERDA.

Quality Assurance Program for BeO-UO_2 Fuel Fabrication

The quality assurance program for the BeO-UO_2 fuel pellet fabrication was finalized. The program was developed in conjunction with Lawrence Livermore Laboratory and includes specifications concerning fabrication, composition, dimensions, density, stacking height, nonconforming parts, and resintering. The program defines the lot definition and identification, survey and monitoring activities, responsibilities for test and inspection, reject fuel pieces, reject lots, records, data packages, in-process inspection and testing, and acceptance requirements.

ACPR Upgrade Test Procedures

The test procedure for the initial criticality test (NST-1) has been approved. A copy of NST-1 has been distributed to personnel involved in the test program. In addition to NST-1, nine other procedures have been written. Hence, the writing of test procedures is one-third completed (there will be a total of thirty test procedures). Table I gives the status of all test procedures. A brief description of the test procedures completed this quarter is presented below.

SOT-5 Electrical Power System

SOT-5 will demonstrate that the AC power distribution system for the ACPR Upgrade console is properly installed and sufficiently sized to reliably provide the required power. This SOT covers all required power distribution for reactor diagnostic instrumentation, reactivity control systems, and the reactor protective systems. It does not cover ventilation cooling or area radiation monitoring systems.

SOT-6 Radiation Monitoring System

SOT-6 is intended to be a functional check of the health physics radiation monitoring instruments. Instruments to be included are the remote area monitors (RAM's) and the constant air monitors (CAM's) associated with the ACPR operation.

TABLE I

Status of ACPR Upgrade Test Procedures

Test	Description	Written	Approved
CT-1	Component Inspection		
CT-2	Cold Function Test		
CT-3	Instrument Calibration and Adjustment		
CT-4	Component Operation		
SOT-1	Reactor Water System		
SOT-2	Reactor Control System		
SOT-3	Plant Protection System		
SOT-4	Ventilation System		
SOT-5	Electrical Power System	Yes	
SOT-6	Radiation Monitoring System	Yes	
SOT-7	Fuel Handling System	Yes	
IPT-1	Integrated Plant Tests		
NST-1	Initial Criticality Test	Yes	Yes
NST-2	Operational Fuel Loading	Yes	
NST-3	Control Rod Worths	Yes	
NST-4	Sample Worth Measurements	Yes	
NST-5	Steady-State Operation	Yes	
APT-1	Rise to Steady-State Power	Yes	
APT-2	Temperature Coefficient	Yes	
APT-3	Coolant System		
APT-4	Nuclear Instrumentation Calibration		
APT-5	Flux Measurements		
APT-6	Flux Spectrum		
APT-7	Normal Pulse Mode		
APT-8	Sample Worth Measurements		
APT-9	Radiation Survey		
APT-10	Cladding Integrity		
APT-11	Reduced Tail Pulse		
APT-12	Multiple Pulse		
APT-13	Repetitive Pulse		

SOT-7 Fuel Handling System

The purpose of the fuel handling system test is to check for proper functioning of the fuel handling facility. In addition, this test will provide fuel handling experience prior to the actual fuel loading.

NST-2 Operational Fuel Loading

The purpose of the operational fuel loading test is to complete the fuel loading process in a safe and efficient manner so as to achieve an operational fuel loading. An operational fuel loading is a loading which is predicted to be adequate to allow the ascension to full power and pulsing test to commence. The fuel loading achieved at the completion of this test should not be considered as the final fuel loading since later adjustments to the core are expected. Reflector elements will also be loaded during this test.

NST-3 Control Rod Worths

The purpose of the control rod worth test is to obtain measurements of the control, transient, and safety rod worths. This test is divided into three phases. The purpose of each phase is as follows:

Phase 1 - The purpose of this phase is to obtain preliminary control, transient, and safety rod worths estimates after achieving initial criticality (NST-1) to assure that NST-2 can commence safely.

Phase 2 - The purpose of this phase is to obtain control, transient, and safety rod worth measurements during the operational fuel loading test (NST-2).

Phase 3 - The purpose of this phase is to obtain control and transient rod worth measurements with various samples in the core cavity (NST-4) to determine the influence of the samples on the transient rod worths.

NST-4 Sample Worth Measurements

The purpose of this test is to determine the reactivity worth of various samples in the irradiation cavity. In addition, this test will be used with NST-3 (Phase 3) to determine the influence of the sample on the transient rod worth.

NST-5 Steady-State Operation

The purpose of this test is to observe and verify reactor and related systems performance during low power steady-state operation. Preliminary calibration of nuclear instrumentation will also be performed during this test.

APT-1 Rise to Steady-State Power

The purpose of this test is to perform an incremental rise in reactor power to full steady-state design power or to maximum allowable fuel temperature. All nuclear startup tests must be satisfactorily completed prior to the test. At each incremental power step the fuel temperature, control rod positions, and operational behavior of the core will be observed. In addition, tests APT-2, -3, -4, -5, -6, -9, and -10 will be performed concurrently with test APT-1. This test (APT-1) will be the controlling test for all of the concurrently performed APT tests.

APT-2 Temperature Coefficient Measurements

This test is required to obtain the temperature coefficient of reactivity and temperature defect data for nontransient conditions. Temperature defect and temperature coefficient measurements will be obtained from the measured fuel temperatures, control rod positions, and control rod worth data at various steady-state power levels.

CHAPTER II

TASK 2. CORE NUCLEAR DESIGN

P. S. Pickard, 5420A; D. J. Sasmor, 5422

Final design calculations have been completed for the core design of the ACPR Upgrade. Fuel loading and uranium enrichment have been finalized. Performance and physics characterization have been completed. Calculated ACPR Upgrade capabilities for reactor safety experiments are being obtained to compare with other test facilities. A brief discussion of final performance estimates and physics parameters are included in this report. Final ACPR Upgrade core specifications are summarized in Table II.

Eigenvalue calculations were performed for the ACPR Upgrade using several different calculational models and cross section sets. Most calculations were performed with the discrete ordinates code TWOTRAN using R-Z geometry. Confirmatory calculations were performed with Monte Carlo codes using both an R-Z geometry and the correct hexagonal lattice. The results of these eigenvalue calculations are summarized in Table III. The spread in the results of the various models is about 1.3 percent. The configurations modeled include the effects of the three-void followed transient rods. The maximum design excess reactivity for the nominal 200-element core is \$8.00 which is equivalent to $k = 1.062$.

Fission density and flux distributions were calculated with TWOTRAN in R-Z geometry. The fission density distribution obtained from these calculations is shown in Figure 2. The S_N calculations show the peak fission density occurring at the cavity fuel interface. Monte Carlo calculations performed to examine this effect indicate that peak at the cavity interface is overestimated by the S_N analysis. The flux distribution resulting from the S_8 calculation is shown in Figure 3.

TABLE II

ACPR Upgrade Design Specifications

Fuel	UO_2 - BeO
	21.5 w/o UO_2 in BeO
	35% Enrichment
Cell Volume Fractions	Fuel - 0.554
	Clad - 0.0392
	Liner - 0.032
	Coolant - 0.2682
Core Geometry	
Cell Area	15.06 cm^2
Lattice	Hexagonal
Pitch	4.171 cm
Number of Elements	200
Fuel Element Diameter	3.747 cm
Homogenized Radial, Inner/Outer	13.319/33.56 cm
Fuel Height	52.25 cm
Control Rods	
Regulating	6
Pulse	3
Safety	2

TABLE III

Eigenvalue Results for Reference ACPR Upgrade
Core Configuration and Final Fuel Composition

Calculational Model	Energy Groups	Eigenvalue	Code
2D - (R-Z), P1, S_4	9	1.0685	TWOTRAN
2D - (R-Z), P1, S_8	9	1.0604	TWOTRAN
2D - (R-Z), P1, S_4	18	1.0626	TWOTRAN
2D - Monte Carlo (R-Z)	9	1.056 ± 0.006	KENO-II
3D - Monte Carlo (Hexagonal)	9	1.07 ± 0.003	MORSE

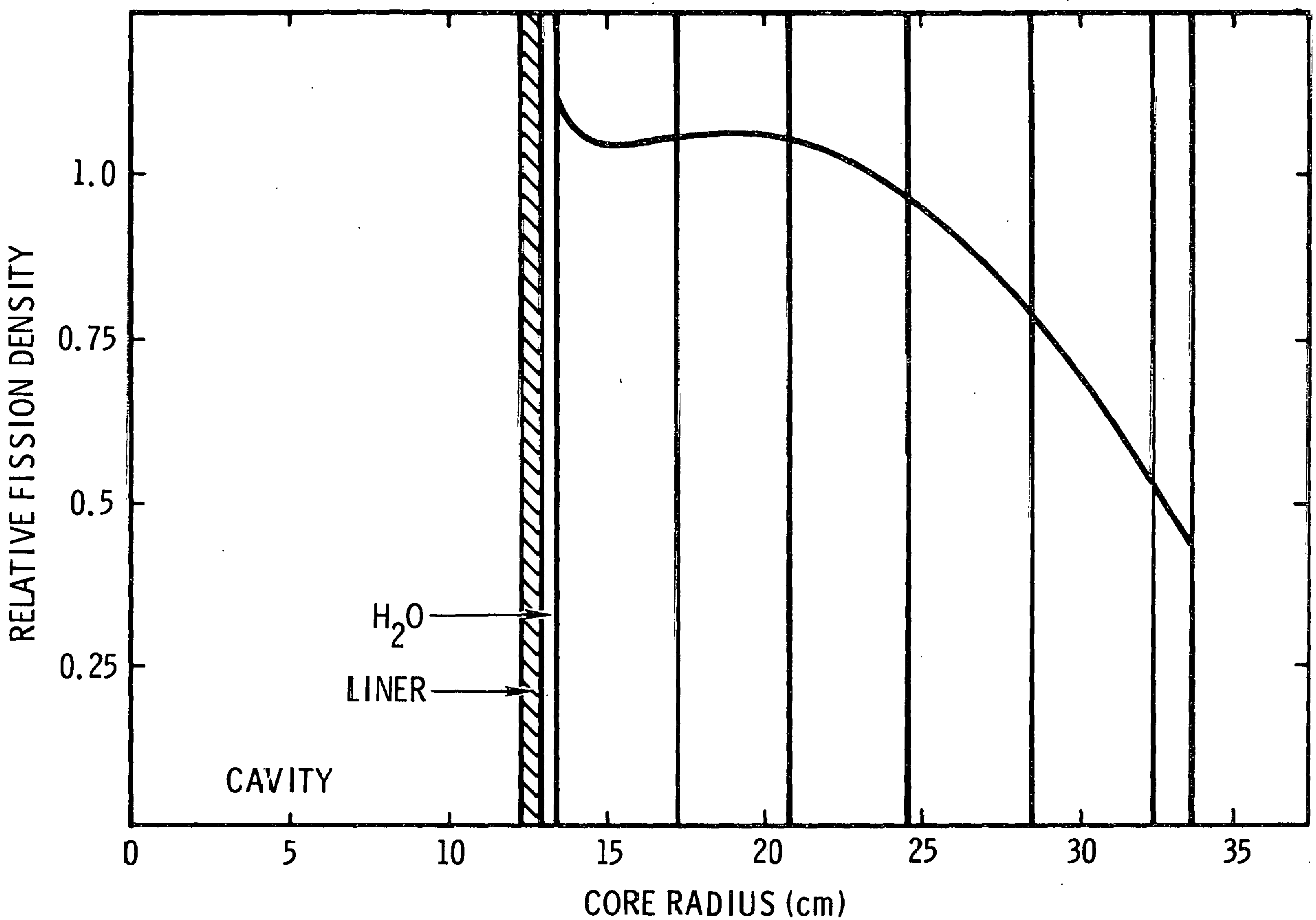

Figure 2. Fission Density Distribution for ACPR Upgrade (Axial Midplane, TWOTRAN, S_8, P_1, 9 Groups)

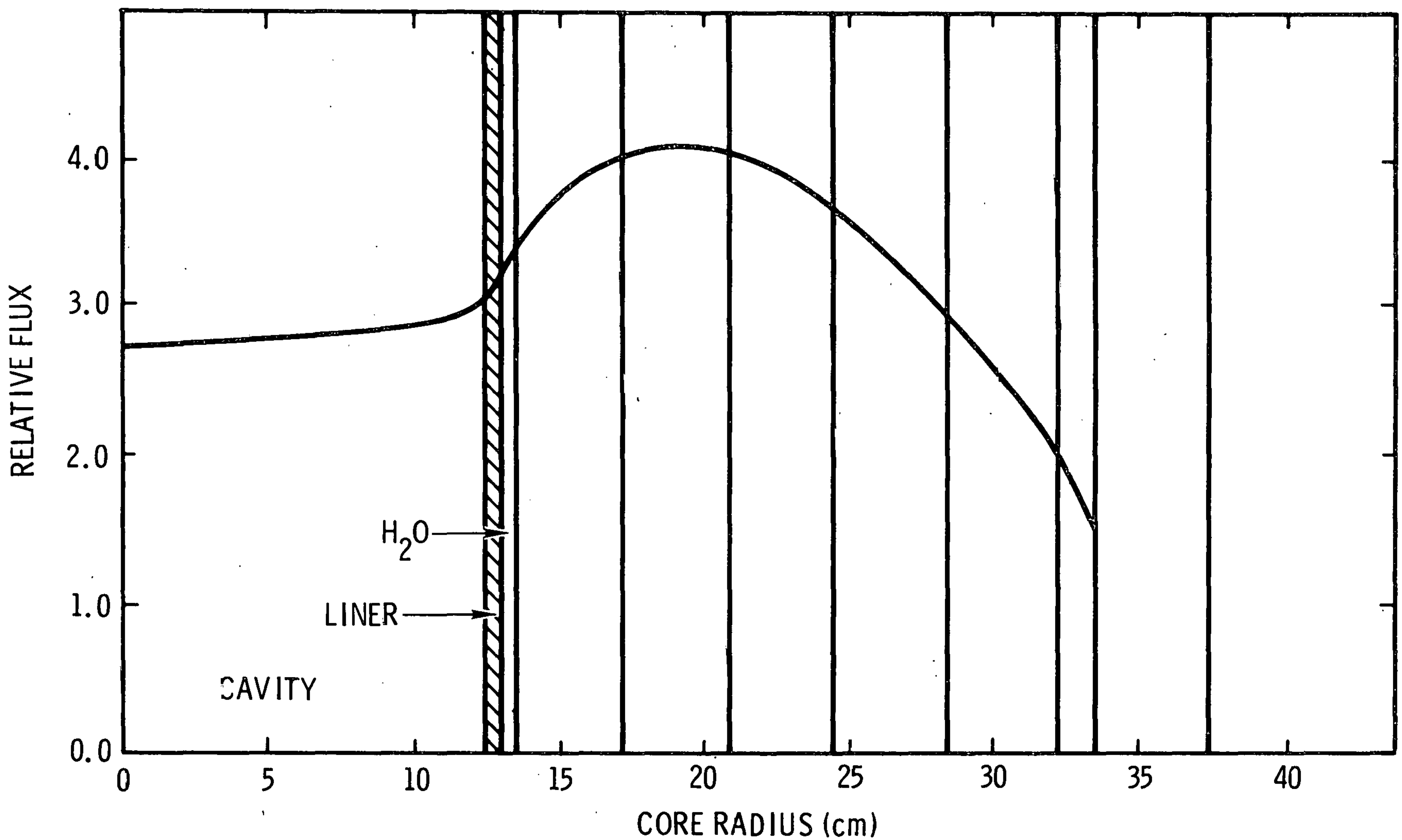

Figure 3. Flux Distribution for ACPR Upgrade
(Axial Midplane, TWOTRAN, S_8, P_1, 9 Groups)

The fission density peak to average factors was calculated with S_8 TWOTRAN analysis and the results are given below.

ACPR Upgrade Peak/ Average Fission Density Factors		
	Overall	- 2.19
	Homogeneous Core	- 1.76
	Axial	- 1.26
	Radial	- 1.40
	Cell	- 1.24

Performance estimates were obtained from TWOTRAN and Monte Carlo analysis. The Monte Carlo and TWOTRAN results agreed within five percent on pulse fluence estimates. For a 300 MW/sec energy release, corresponding to an adiabatic peak temperature of 1400°C, the estimate of total fluence is 6.5×10^{15} n/cm^2.

Two approaches were utilized in determining the prompt negative temperature coefficient. TWOTRAN was used with two different isothermal cross section sets to arrive at a reactivity loss with temperature. The average temperature coefficient determined from this approach for 27 - 627°C was -0.45 ¢/°C. The 1D space time kinetics code SAK was also used to estimate temperature effects. The comparable results from SAK, using distributed temperatures from a cross section interpolation table was -0.49 ¢/°C. Calculations indicate that approximately sixty percent of the temperature feedback is due to Doppler broadening of U^{238} resonances, the remainder being due to spectral hardening caused by the prompt heating of the BeO matrix material.

The neutron generation time was calculated by Monte Carlo (KENO-II) to be 24 μs. Placing a representative amount of moderating material in the cavity increases the neutron generation time to 27 μs. Removal of the outer nickel reflecting elements increases the generation time to 35 μs.

The neutron energy spectrum as calculated with an eighteen-group cross section set in TWOTRAN is shown in Figure 4. The calculated fraction of neutrons greater than 10 KeV for the Upgrade is 0.58. The comparative number for the current ACPR is 0.52. The dynamic response of the ACPR

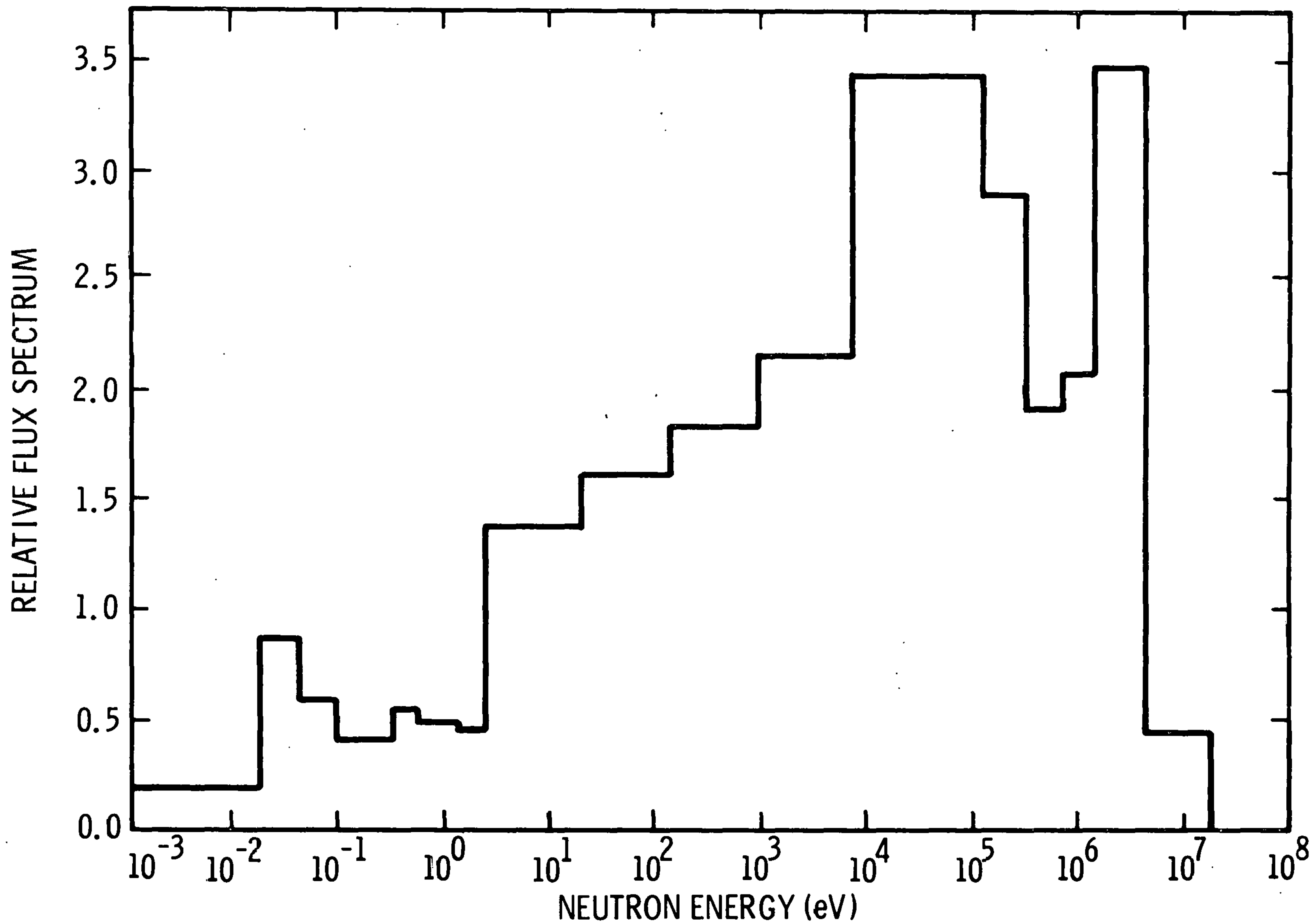

Figure 4. Cavity Flux Spectrum in ACPR Upgrade (UO_2-BeO)

Upgrade has been analyzed using both point kinetics and space-time diffusion codes. Both codes include coupled heat transfer for representative fuel elements in the core. Pulse shapes and initial risetimes obtained by the two methods agree well for the same description of temperature feedback. A design pulse calculated with space-time diffusion analysis is shown in Figure 5. A more detailed summary of kinetics studies will be included in the next quarterly report along with predicted ACPR Upgrade performance capabilities with reactor safety experiments.

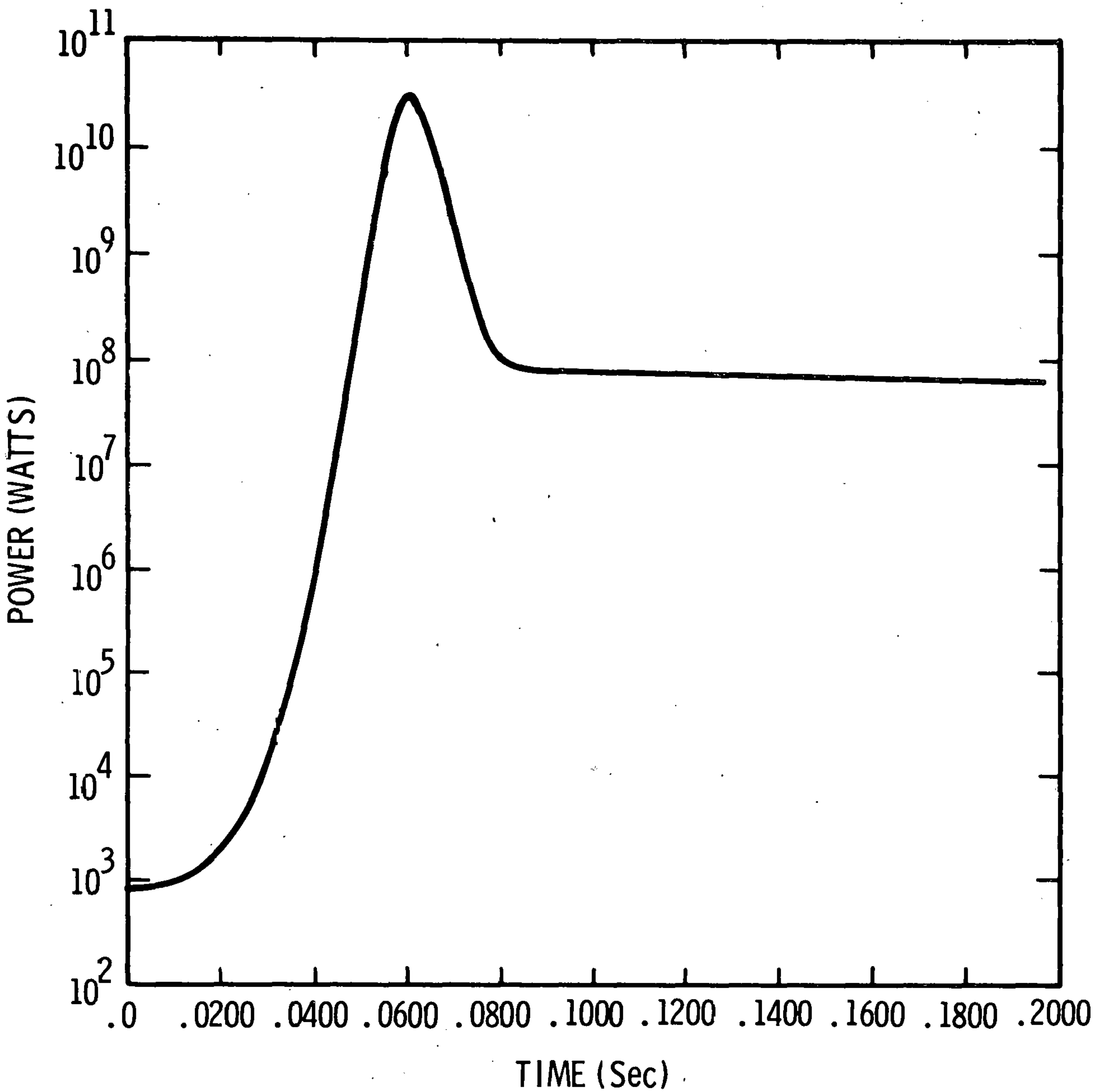

Figure 5. Single Pulse Power Trace for Reactivity Insertion of \$2.6 (1D Space-Time - 5 Groups)

CHAPTER III

TASK 3. ACPR UPGRADE CONSOLE DEVELOPMENT

W. H. Sullivan, 5423

Design, prototype fabrication, evaluation and design review of the percent power, NV/NVT, thermocouple calibrate circuit, and the rhodium NVT systems were completed. Design of the rod drive electronics, rod position readout, and the control system diagnostic instrumentation drawer was also completed. The entire system design is now approximately 75 percent complete. A satisfactory preliminary design review of these parts of the system was conducted.

Mechanical assembly and electrical wiring of the log power, linear power, shutdown, and magnet power supply drawers are in progress. The preliminary design review of the protect system and the panel layouts for the entire control console was finished.

CHAPTER IV

TASK 4. MECHANICAL DESIGN

H. C. Walling, 1136

This quarter discussions were begun with EG&G of Idaho Falls concerning the installation of the mechanical equipment inside the ACPR tank. Detail drawings are in preparation and include a description of the assembly procedure.

Fans and filters are being fabricated for the reactor cavity ventilation. Fabrication of the experiment storage facility is proceeding on schedule; modifications have been made to seal the facility properly.

The Penthouse is scheduled for installation on October 1 and the contractor has erected the steel structure for this installation.

Blowdown and filtered flow path alterations for the neutron radiography facility were defined and detail design is now in progress.

Electrical cable runways were designed to separate and absolutely protect, from fire or physical abuse, PPS and reactor control instrument lines.

CHAPTER V

TASK 5. FUEL ELEMENT DESIGN

J. A. Reuscher, 5450

Introduction

This task involves the design and testing of the $BeO-UO_2$ fuel elements. The results of the in-pile fuel experiments were used to develop the fuel material and the fuel dual-annulus configuration.

Fuel Element Design

The details of the BeO_UO_2 fuel element design were given in the last quarterly report.[10] Since the design was final, there was no further activity on the fuel element design during this quarter. The non-fuel components (cladding, niobium cups, and BeO end pieces) were ordered. The design of the fuel-followed control rod was completed and a preliminary design for the instrumented fuel element was begun.

CHAPTER VI

TASK 6. FUEL ELEMENT FABRICATION
C. H. Karnes, 5835

The fuel fabrication schedule devised in January called for LLL to begin fabrication of U^{235} fuel on May 9 and LASL to begin fuel element assembly on August 4. That schedule had considerable planned slack time which has now been eliminated because of a variety of problems with the fuel fabrication process.

The fuel fabrication process involves the following steps:

1. Dissolve uranyl nitrate in deionized water
2. Filter to remove insoluble impurities
3. Slurry with proper amount of BeO
4. Precipitate uranium compound with addition of ammonium hydroxide
5. Filter and dry
6. Calcine at 1000°C in air to produce U_3O_8/BeO cake
7. Mill to break up cake
8. Add poly (ethyleneglycol) binder in V-blender
9. Blend until proper granulation has occurred resulting in optimum powder flowing characteristics
10. Press pellets in automatic press
11. Sinter pellets at 1750°C in hydrogen which reduces U_3O_8 to UO_2
12. Inspect for density and dimensions
13. Pack and ship to LASL for assembly

The problems with the powder flowing characteristics result from Steps 7 and 9 which determine the particle size distribution and density of individual particles. These properties determine the bulk powder density as the pressing die is filled in Step 10, the uniformity of powder flow within the die, and the uniformity of green pellet densities. The green pellet density determines the amount of shrinkage during sintering and, hence, the final pellet size. Steps have been taken to improve the powder flowing characteristics, including contingency powder granulation processes, and it is anticipated that a solution to the problem is imminent.

Part of the slip in the schedule has been the result of delayed delivery and poor quality of the tool steel dies which were to be used until the tungsten carbide dies were received. It appears now that the carbide dies will arrive before pressing of enriched fuel begins.

The fuel quality assurance program has been completed. It includes the following tests and inspections:

1. Chemical analysis for UO_2 content (21.5 ± 0.5 w/o)
2. Isotopic analysis for enrichment (35.0 ± 0.2% U^{235})
3. One hundred percent dimensional inspection using an optical comparator. This may be relaxed to a sampling basis if justified by the early history.
4. Visual inspection for chips and cracks
5. A 10-percent random-sampling for immersion density measurements
6. The stacking height of 16 randomly selected pellets shall not exceed 4.025 in.

The rejection rate is unknown but a rate of less than 10 percent is the objective.

The data recording system has been devised for the analyses and inspection reports to be included with each mix of fuel. A mix is one day's production. The data package will stay with the fuel until it is assembled into a fuel rod at LASL at which time appropriate data from it will be transferred to each fuel rod data package.

In order that LASL can assemble a prototype fuel rod, a sufficient quantity of pellets made with U^{238} was shipped. The other components necessary to assemble a prototype fuel rod have also been delivered to LASL.

The problems experienced during this quarter are well under way to being solved and a new fabrication schedule is being formulated to insure that no critical date in the program will be slipped. The schedule will include all necessary overtime at both LLL and LASL.

CHAPTERS VII and VIII

TASKS 7 and 8.

These tasks are no longer a part of this project (see Introduction of this report for explanations).

CHAPTER IX

TASK 9. DIAGNOSTIC SYSTEM

Activities on this task for this reporting period are described in the Fast Reactor Safety Research Program - Quarterly Report, April - June 1977, SAND77-1134, Advanced Reactor Research Department, Sandia Laboratories, Albuquerque, New Mexico, September 1977.

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