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Annular Core Pulse Reactor Upgrade
Quarterly Report
October-December 1976

Reactor Research and Development 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 March 1977



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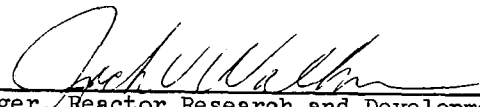
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ANNULAR CORE PULSE REACTOR UPGRADE QUARTERLY REPORT*
October - December 1976

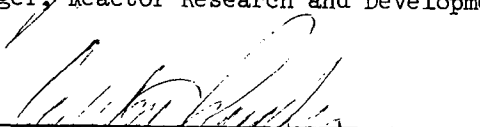
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Submitted by
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Approved:



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*This work was supported by the U. S. Nuclear Regulatory Commission Project No. A1032 and the U. S. Energy Research and Development Administration Under Contract AT(29-1)-789.

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ANNULAR CORE PULSE REACTOR UPGRADE QUARTERLY REPORT

INTRODUCTION

The objective 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 increased degradation of the pulse duration. The upgraded reactor will also have an increased steady-state neutron flux. The original approach to the upgrade modification involved a two-region core concept. The inner region, surrounding the irradiation cavity, would consist of a high-heat-capacity fuel which would sustain a large fission energy deposition. The outer region would consist of a uranium-zirconium hydride fuel similar to the present ACPR fuel. This reactor modification will make use of the majority of the existing reactor structure. At the end of this reporting period the decision was made to go to a single region high-heat-capacity fuel core. This design will be discussed in more detail in the next quarter's report.

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.⁴⁻⁸

The ACPR Upgrade project is divided into nine tasks to improve management of the overall project and to maintain close control of the project budget. 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 Safety, Standards and Compliance 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 containment structure, 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 development tasks (Tasks 6 and 7) and includes the stress analysis and heat transfer studies for design of the high-heat-capacity fuel elements. The fuel element demonstration tests are also a part of this task.

Task 6. Primary Fuel Material Studies (R. L. Coats, Supervisor)

The primary fuel material (at the present time) is BeO-UO_2 since it offers the largest performance improvement for the upgrade. This task involves the development of fabrication techniques, material compatibility studies, material property determinations, material analysis, and in-pile experiments for pulse testing of fuel geometries.

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

The secondary fuel material is (UC-ZrC)-graphite which will be used in the high-heat-capacity fuel element if the BeO-UO₂ does not prove feasible. This task involves the development of fabrication techniques, material compatibility studies, material property determinations, material analyses, and in-pile experiments.

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

The testing of the outer core fuel material and the design of the driver core fuel element are the objectives of this task. This fuel is a uranium-zirconium hydride which is similar to the present ACPR fuel. The hydrogen-to-zirconium ratio is decreased slightly for the upgraded reactor. The testing of this fuel includes pellet tests and prototype element tests.

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. 9).

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.

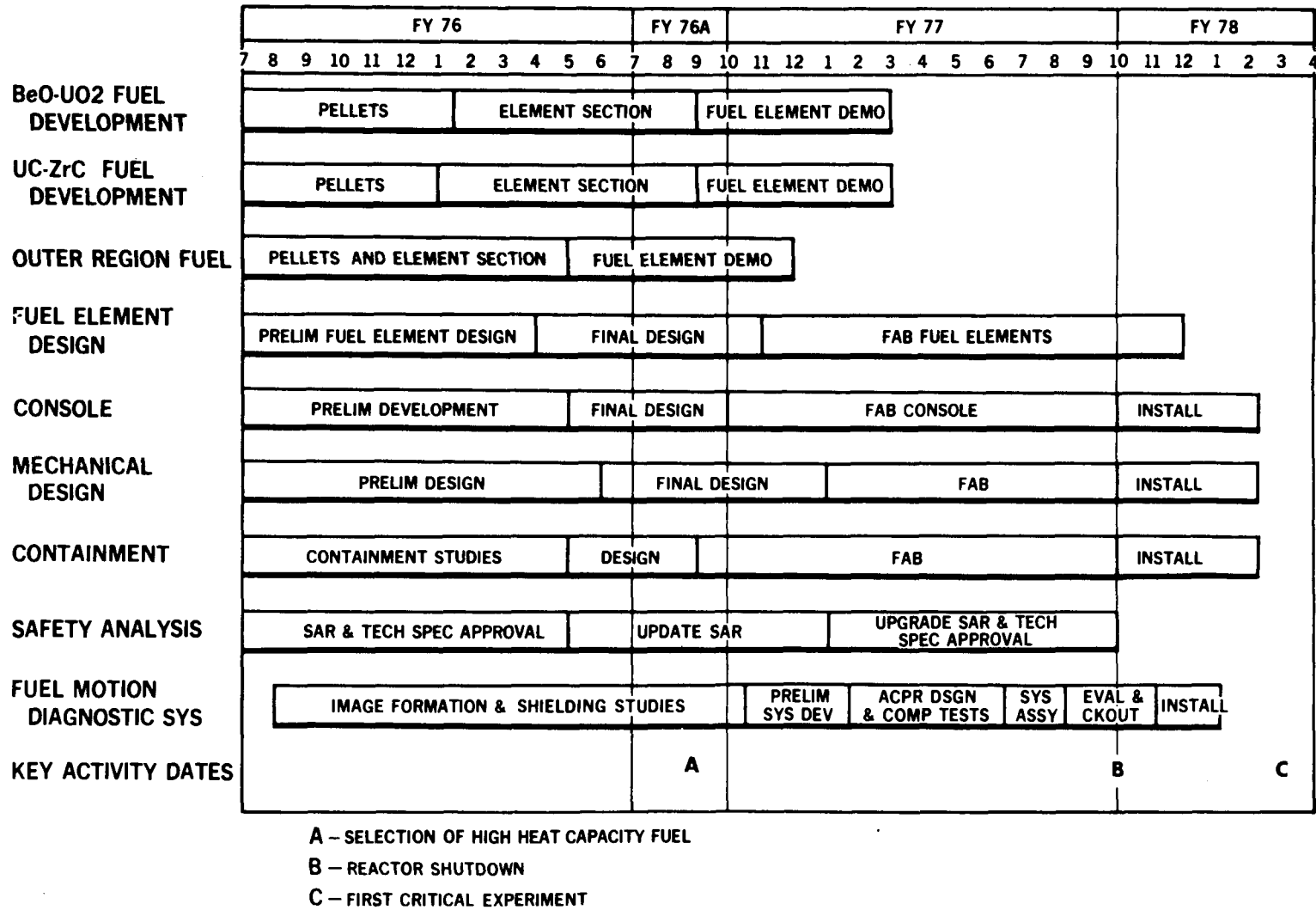


Figure 1. ACPR Upgrade Schedule

CHAPTER I

TASK 1. SAFETY, COMPLIANCE, AND DOCUMENTATION
J. A. Reuscher, 5424; H. C. Walling, 1136;
A. C. Marshall, 5424

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 for the initial tests and operations.

Safety Analysis Report (SAR)

During this quarter, three chapters of the Safety Analysis Report were reviewed by the ACPR Committee. These chapters are Chapter 5, Reactor Coolant System and Converted Systems; Chapter 6, Engineered Safety Features; and Chapter 16, Initial Tests and Operations. Initial drafts of Chapter 4, Reactor; Chapter 7, Instrumentation and Control; Chapter 15, Quality Assurance; and Chapter 14, Accident Analysis were completed.

Accident Analysis

The accident analysis for the ACPR Upgrade with a (UC-ZrC)-graphite central region was completed by Intermountain Technologies, Inc. A total of 29 different kinds of accidents were considered, which included the general categories of:

1. Increase in Heat Removal of the Bulk Coolant System
2. Decrease in Heat Removal by the Bulk Coolant System
3. Decrease in Reactor Coolant System Flow Rate
4. Reactivity and Power Distribution Anomalies
5. Increase in Reactor Coolant Inventory
6. Decrease in Reactor Coolant Inventory

7. Radioactive Release from a Subsystem of Component
8. Anticipated Transients without SCRAM
9. Miscellaneous Initiating Events

Since BeO-UO₂ was chosen as the high-heat-capacity fuel during this quarter (see Chapter V), these accidents were reanalyzed for the BeO-UO₂, and the accident chapter was revised to reflect the fuel choice.

Initial Tests and Operations

The initial test program is divided into five major phases:

1. Component Tests
2. System Operational Tests
3. Integrated Plant Tests
4. Nuclear Startup Tests
5. Ascension to Full Power and Pulsing Tests

The first three are performed prior to fuel loading and are considered to be pre-nuclear tests. These tests will insure that all reactor systems are functioning properly and ready for the nuclear tests. The last two phases include the critical experiment, reactivity calibrations, and the approach to design pulse and steady-state operation.

Clad Rupture Test

An explosive experiment was performed to examine the effect of the rupture of the center element in a seven-pin array of elements. This test simulated the mechanical effect of a waterlogged fuel-element accident. The fuel element clad was constructed to the same specifications as the elements in the ACPR. The clad is a rolled and welded type 304 stainless-steel tube. The wall thickness as designed was 0.05 cm and the outside diameter was 3.73 cm. The measured outside diameter varied from 3.75 to 3.78 cm. The triflute was E-beam welded to the bottom of the clad tube. A special closure, designed to accommodate the pressure transducer, was E-beam welded to the top of the clad tube. Figure 2 is a sketch of the fuel element clad. Seven of these fuel pin clads were constructed, since this was the number required to model one hexagonal cell of the ACPR core.

In order to model the hexagonal array, two support plates were constructed from aluminum alloy 6061-T6. The support plates were held apart with steel threaded rod in four places. A requirement of the test was to rupture the clad under water. The hydrostatic pressure acting on a fuel clad in the ACPR is approximately 23 psia (corresponds to 7.6 metres water column + atmospheric pressure). To meet this requirement, a pressure vessel was constructed so that the entire array would be under water.

The pressure to rupture the clad was produced by burning FFFG black powder inside the closed fuel element. The black powder was initiated by a spark from an electric match. The pressure produced by the burning black powder was recorded with a Kulite HKS-375-10000 (diaphragm type) transducer. Pressure of the gas above the water line inside the containment vessel was recorded with a Kulite XTS-190-300 diaphragm type transducer. This measurement was made in the event the clad ruptured and the explosive products vented to the containment vessel interior. Table I gives physical characteristics of the test apparatus.

TABLE I

Physical Characteristics of Clad Rupture Test Apparatus

Clad OD (cm)-----	3.73
Clad Wall Thickness (cm)-----	.05
Clad Interior Volume (cc)-----	578
Clad Material-----	304 stainless
Fuel Cell Pattern-----	hexagonal
Pitch-to-Diameter Ratio-----	1.10
Containment Vessel ID (cm)-----	20.3
Containment Vessel Internal Volume (cc)	4.42×10^4
Water Volume (cc)-----	1.96×10^4

A number of calibration tests were performed to determine the pressure of explosive product gas versus initial mass. These calibrations were performed inside a fuel clad identical to that shown in Figure 2. The calibrations were performed in the containment vessel; in fact, the only difference between calibration and actual test was that the six clads surrounding the central clad were not present. Figure 3

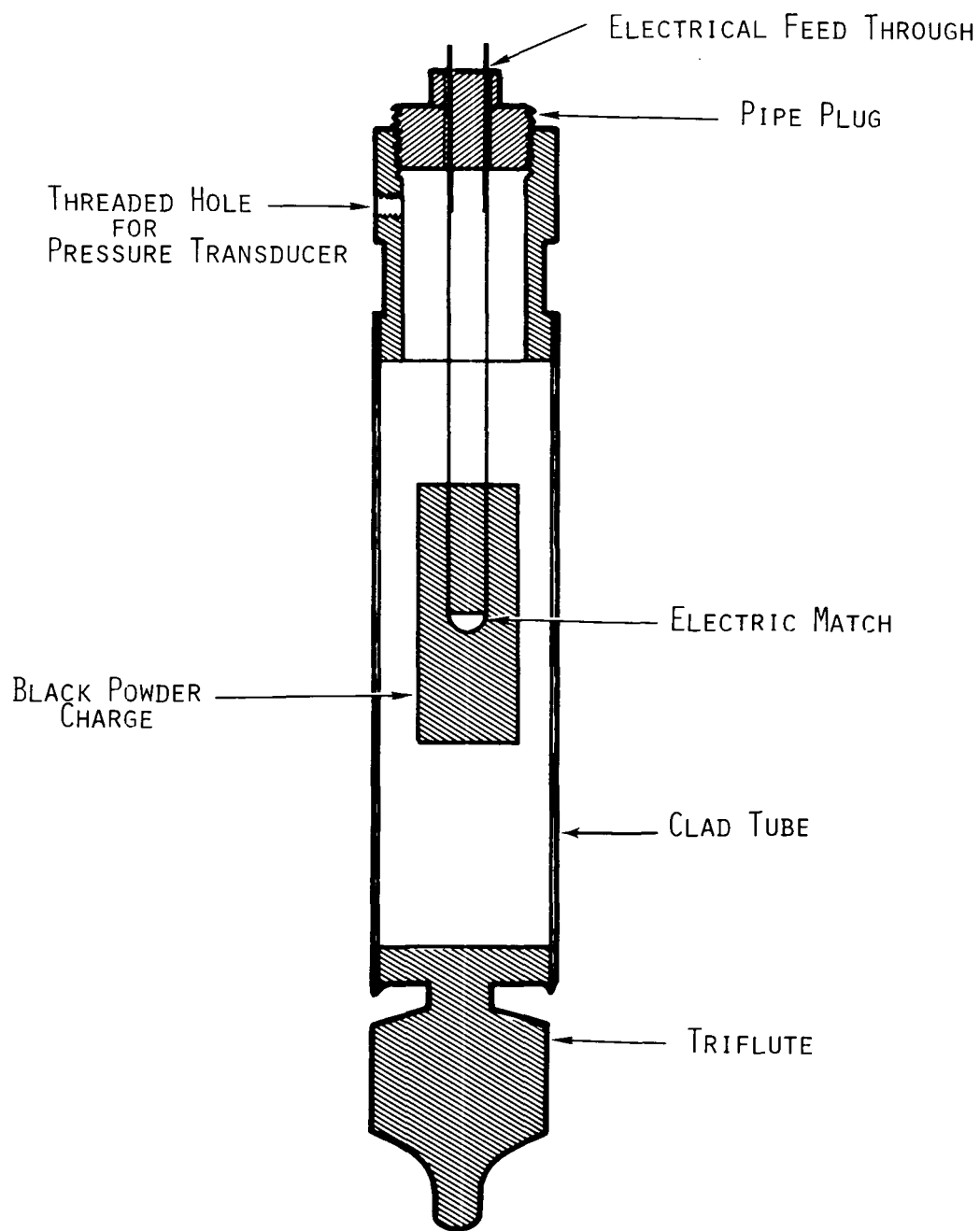


Figure 2. Stainless Steel Clad with Explosive Charge

is a plot of internal pressure versus black powder weight. The single clad tests showed rupture pressures from 2300 to 2900 psi.

The test on the seven-clad array was performed with 50 grams of black powder in the central clad to overcome the lack of repeatability shown in Figure 3. The clad ruptured at about 2330 psi, which produced a pressure rise in the water container of about 72 psi. Examination of the claddings after the test revealed the central clad had split axially over a distance of about 18 cm. The surrounding units were bent and deformed approximately 0.5 cm outward from the center. The outer clads were not fractured.

These experiments have shown that the clad of adjacent elements would not be damaged in the event of a waterlogged fuel-element accident. Only the fission products from the ruptured element need to be considered as the source term for atmospheric transport in the accident analysis.

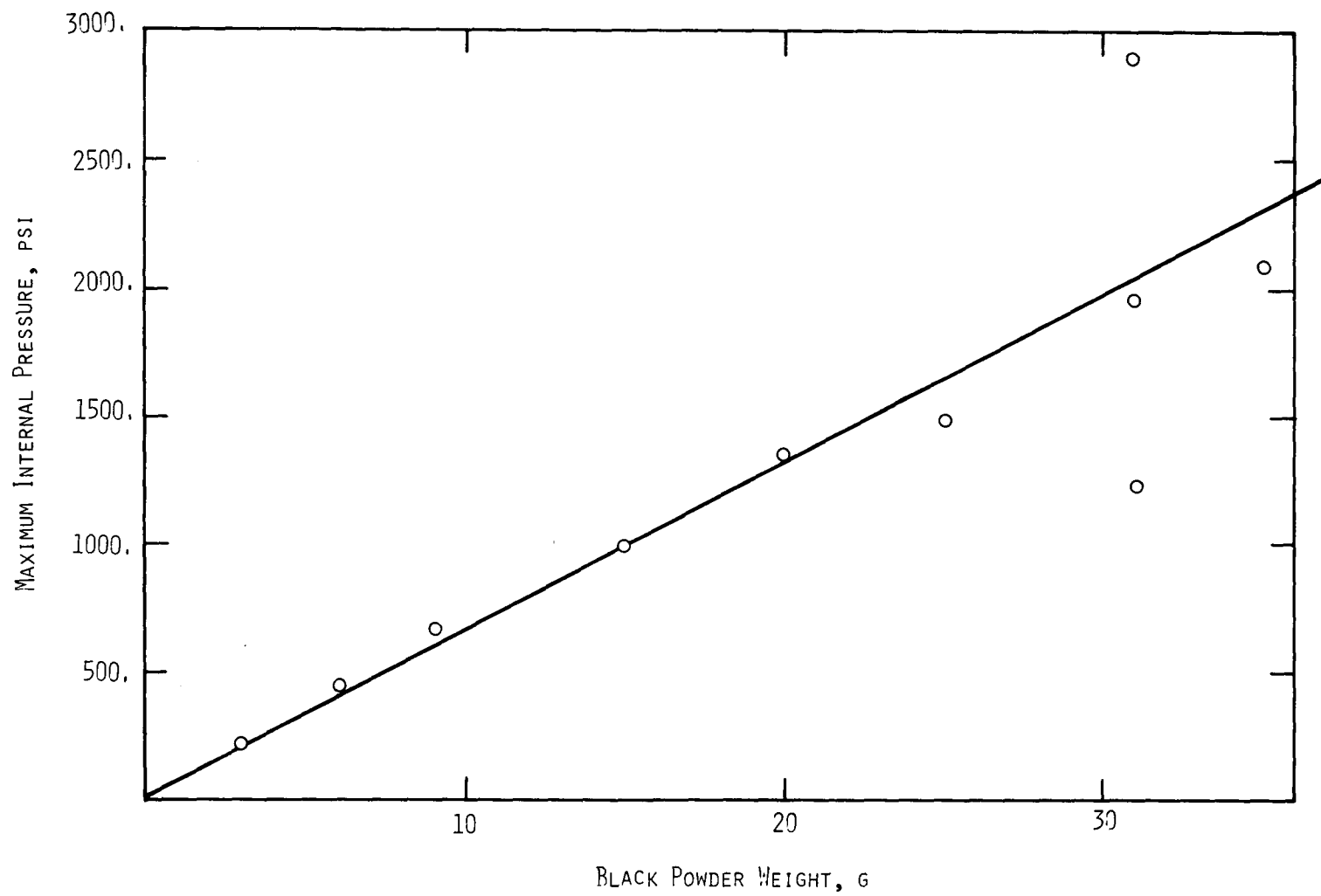


Figure 3. Clad Internal Pressure versus Black Powder Weight

CHAPTER II

TASK 2. CORE NUCLEAR DESIGN P. S. Pickard, 5424; D. J. Sasmor, 5422

The core design calculations for the ACPR Upgrade two-region core were completed for both the UO_2 -BeO and UC-ZrC-C fuel candidates. These calculations were used as a basis of comparison of the performance and kinetic characteristics of these two inner region fuels. A summary of these calculations is included in this report. On the basis of these results and the results of heat transfer studies and in-pile material tests, it was concluded that the UO_2 -BeO was the most satisfactory fuel for the ACPR Upgrade.

The comparison calculations for the UO_2 -BeO and UC-ZrC-C fuels were based on a U-ZrH_{1.5} fuel performance of a maximum temperature rise of 1000°C. The in-pile testing program for the U-ZrH_{1.5} fuel was being conducted in parallel with the core design studies. Early results from these tests indicated that the behavior of the U-ZrH_{1.5} as fabricated was sufficiently different from that of the original hydride fuel to merit reevaluation of the U-ZrH_{1.5} fuel for use in the ACPR Upgrade. This situation was apparently due to changes in manufacturing techniques for this fuel. Due to the uncertainty in the time required to resolve the U-ZrH_{1.5} in-pile behavior and to produce fuel with the original specifications, single-region core design studies were undertaken utilizing UO_2 -BeO as the only fuel. The purpose of these calculations was to compare the performance and kinetic characteristics of the one- and two-region core designs. The elimination of the interface problem in the single-region core allows higher ultimate performance potential for a given maximum fuel temperature. Due to the smaller negative temperature coefficient of the nonhydride fuels, the minimum initial period is generally greater in the single-region core designs, but the penalty can be reduced sufficiently to fit within the overall Upgrade objectives. A summary of the preliminary single-region core calculations is also included in this report.

Two-Region Core Design Studies

The two-region core design approach utilized two fuel types: an inner fuel with high heat capacity and temperature capability, and an outer hydride region. Two fuel candidates were developed for the inner fuel region ($\text{UO}_2\text{-BeO}$ and UC-ZrC-C). The advantages of this approach were the strong negative temperature coefficient and the short initial risetime attainable. The disadvantage with this design was the temperature peaking problem at the two-region interface, which limits pulse performance and induces significant thermal stress in the fuel. This approach does allow considerable design flexibility in that, if the inner-region fuels were to be limited in performance, the worth of the inner-region fuel can be adjusted to accommodate such limitations with minimal impact on overall core performance. The importance of this flexibility has been diminished by the successful development of the $\text{UO}_2\text{-BeO}$ and UC-ZrC-C fuels.

The calculational approach used in the two-region core studies is summarized in Table II. The primary physics quantities of interest were the minimum initial period and the pulse fluence in a maximum pulse. Several reference core configurations were defined and these quantities evaluated. The standard core geometry used was three rows of high-heat-capacity (HHC) fuel and three rows of $\text{U-ZrH}_{1.5}$ fuel. The third row of HHC fuel is actually a buffer zone with reduced loading to control the fission density peaking at the interface of the two fuel types. The fuel enrichment was varied in these core designs to provide a means of varying the pulse width with a small penalty in pulse performance. The fuel loading and enrichment for the reference cores evaluated are shown in Table III. Fuel loadings did not exceed the amounts that were considered maximum for satisfactory mechanical properties in the fuel materials.

The primary quantities of interest in comparing the $\text{UO}_2\text{-BeO}$ and the UC-ZrC-C fuels in a multi-region core were:

1. pulse fluence improvement
2. fuel temperature and stress
3. kinetic characteristics (temperature coefficient, generation time, pulse width)
4. steady-state flux and fuel temperature
5. neutron energy spectrum

TABLE II

Core Design Computational Approach

1. Survey Calculations	--	1D - S_N - DTF	
		2D - S_N - TWOTRAN	
2. Reference Core Calculation	--	2D - S_N - TWOTRAN	
3. Peaking Factors	--	3D - Monte Carlo - MORSE	
		2D - S_N - TWOTRAN	
4. Kinetics Calculations	--	A) Reactivity Loss	-- TWOTRAN
		Generation Time	-- KENO, MORSE
		B) ID, Time-Dependent	
		Diffusion with	-- SAK
		Heat Transfer	
5. Cross Sections	--	ENDF/B-IV	-- AMPX
		ENDF Processor	-- XLACS
		Resonance Calculation	-- NITAWL
		ID, S_N Weighting	-- XSDRN

TABLE III

Reference Core Fuel Loading and Enrichment

Core	<u>BeO-UO₂</u>	
	Inner Region (w/o UO ₂)/(Enrichment)	Outer Region (w/o U)/(Enrichment)
B1	26 w/o / 45 e/o	12 w/o / 10 e/o
B2	17 w/o / 65 e/o	11.3 w/o / 10 e/o
B3	10 w/o / 93 e/o	11 w/o / 10 e/o
B4	10 w/o / 93 e/o	5.2 w/o / 20 e/o
Core	<u>UC-ZrC-C</u>	
	Inner Region (mg/cc U)/(Enrichment)	Outer Region (w/o U)/(Enrichment)
C1	750 mg/cc / 60 e/o	12.2 w/o / 10 e/o
C2	400 mg/cc / 93 e/o	11.5 w/o / 10 e/o
C3	350 mg/cc / 93 e/o	5.4 w/o / 20 e/o

The fission density profiles are qualitatively similar for both fuels, and a representative fission density distribution is shown in Figure 4 for a UC-ZrC-C core. Rows 1 and 2 are full loading with Row 3 reduced to approximately half loading to accommodate fission density peaking due to the outer hydride region. The peak fission densities are adjusted to take advantage of the volumetric heat capacities of each fuel. The outer reflector in these calculations is water. The strong peaking effects in Rows 2 and 3 result in high peak temperatures in the interface region which ultimately limit pulse performance.

A second consideration involving the interface region is the large temperature drop across individual fuel pellets. The thermal stress generated in this region was evaluated for several reference core configurations. A MORSE Monte Carlo calculation was used to provide a temperature profile across the cylindrical fuel pellet. These adiabatic temperature profiles were used in a two-dimensional finite element stress code (SASL) to determine maximum stress in the fuel pellet. These stress magnitudes were compared with those calculated for in-pile fuel-pellet tests. The maximum stresses occurring in the Upgrade core were generally about 20 percent lower than those calculated for the most severe in-pile tests.

For purposes of comparison of the BeO and graphite based fuels, it is sufficient to compare the peak-to-center values calculated with Monte Carlo. A summary of these results is given in Figure 5. In general, temperature profiles are somewhat more severe in the UC-ZrC-C core than in the UO_2 -BeO. This is due to the higher fissile loading required in the UC-ZrC-C fuel to obtain a given reactivity worth. These fission density profiles (P/C) calculated for the in-pile tests ranged from 2.0 to 2.5 for the various moderator thicknesses used, while measured values ranged from approximately 2.1 to 2.3.

The neutron energy spectrum for the Upgrade core is significantly harder for either the UO_2 -BeO or the UC-ZrC-C fuels. The higher fissile loading required in the UC-ZrC-C core results in a slightly harder cavity spectrum than the UO_2 -BeO core. A comparison of the cavity flux spectrum for both Upgrade fuels and the current ACPR is shown in Figure 6. These results are from two-dimensional S_N calculations using nine group cross sections.

The fraction of neutrons above 10 keV is plotted as a function of radius for the UO_2 -BeO core in Figure 7. The UC-ZrC-C core and current ACPR cavity fraction

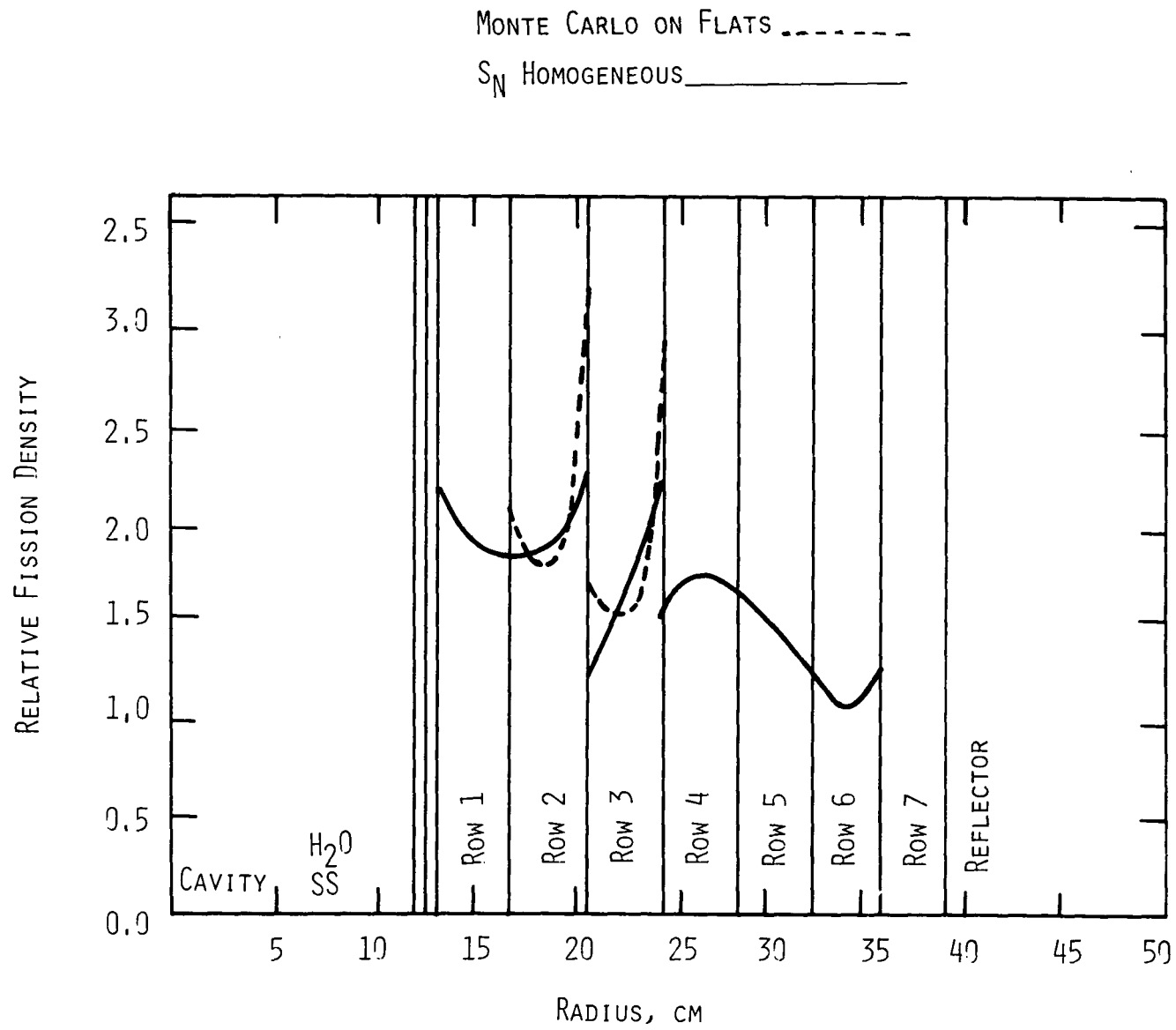
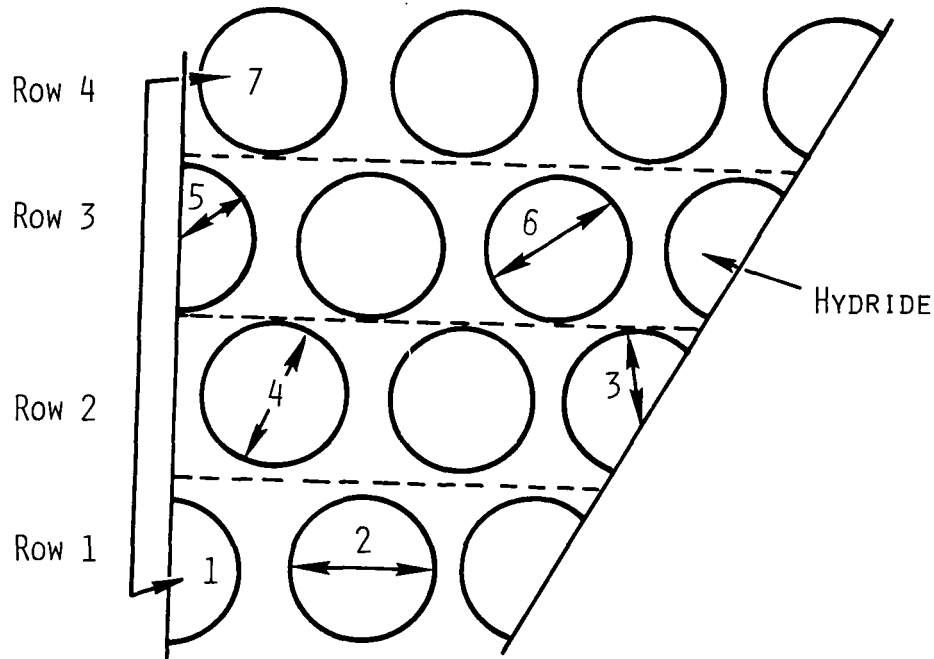


Figure 4. Fission Density Peaking in UC-ZrC-C Core Calculated by S_N and Monte Carlo



DESCRIPTION	REF. CORE #C1 - CARBIDE				REF. CORE #B2 - BeO		
	ELEMENT	P/A	P/C	E/E ₀	P/A	P/C	E/E ₀
R3 NEXT TO HYDRIDE	6	1.58	1.88	-	1.52	1.63	-
R3 ON FLAT	5	1.52	1.96	-	1.45	1.85	-
R2 ON FLAT	4	1.69	1.96	-	1.62	1.72	-
R2 NEXT TO HYDRIDE	3	1.78	2.17	-	1.57	1.88	-
MISPLACED R1-R4	(1→7) 7	1.68	2.03	1.66	1.67	1.87	1.34
MISPLACED R4-R1	(7→1) 2	1.8	1.98	.998	1.89	2.12	.993

P/A VALUES ±5%

P/A VALUES ±7%

Figure 5. Fission Density Profiles in BeO and Graphite Two-Region Core Designs

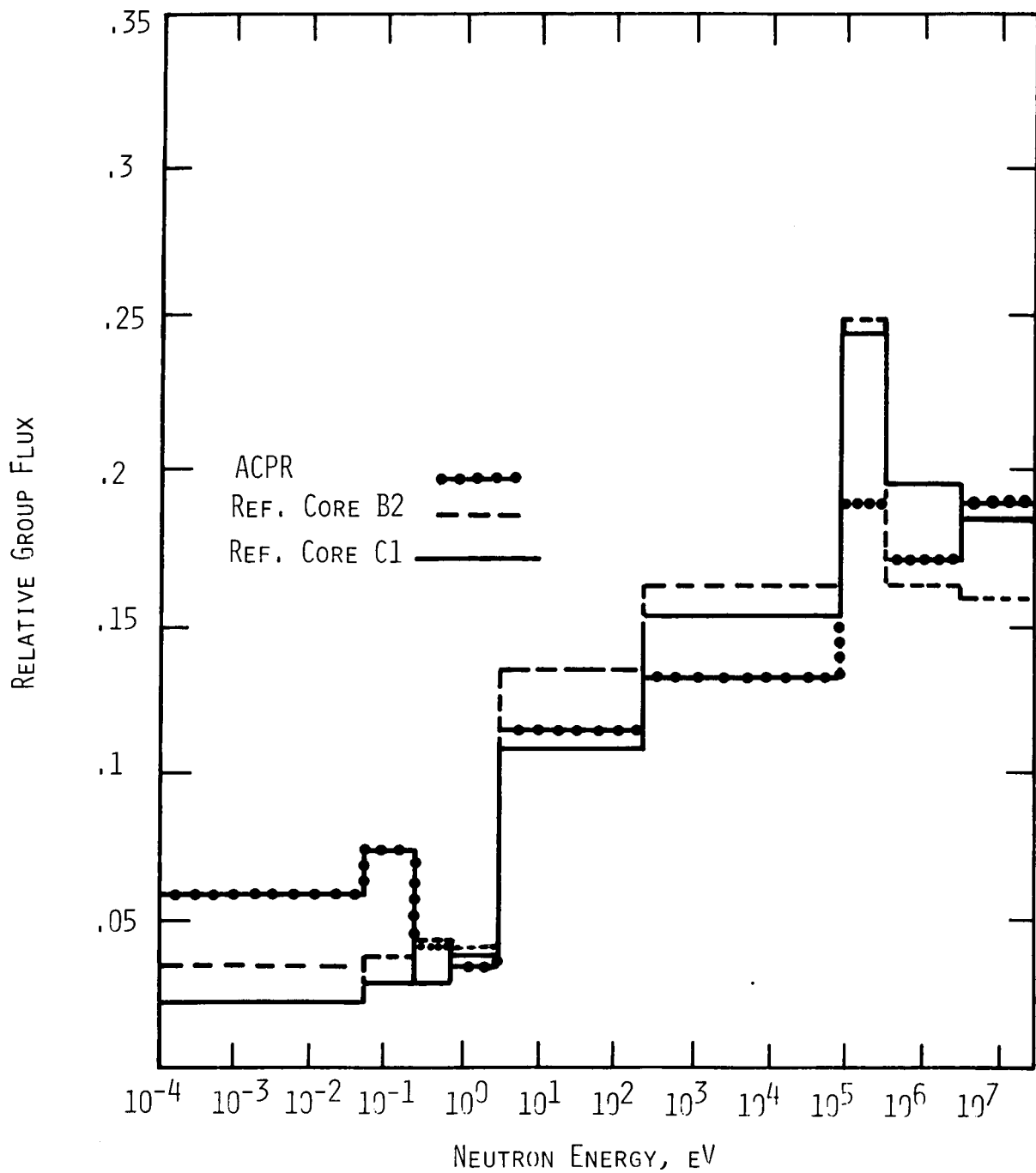


Figure 6. Cavity Flux Spectrum for Two-Region Core Designs

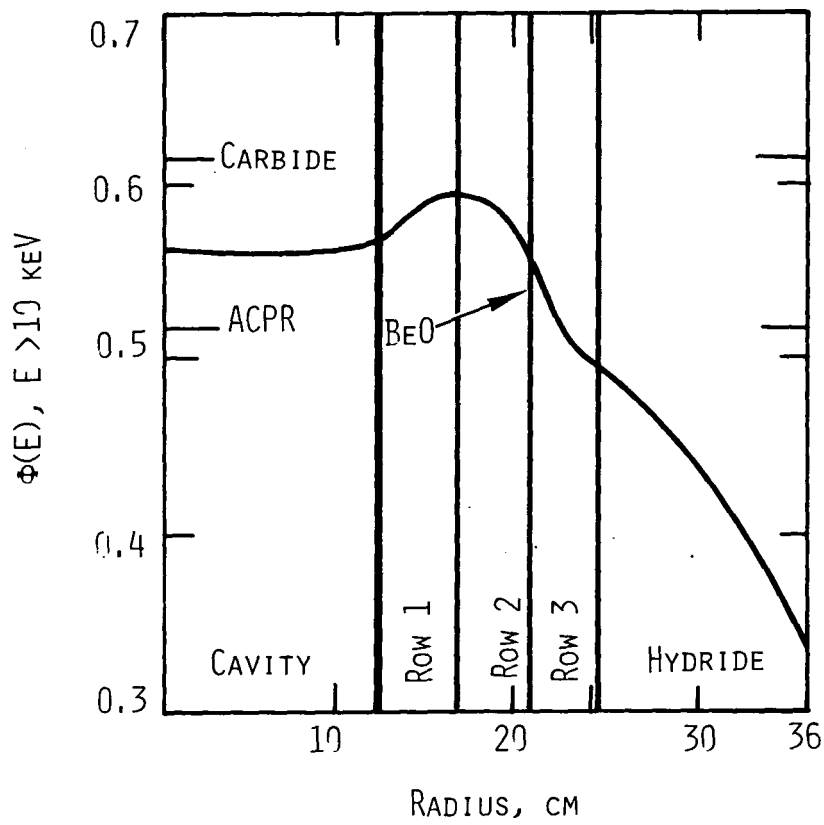


Figure 7. Fraction of Neutrons Greater than 10 keV

are shown for comparison. The neutron energy spectrum hardens significantly in the high-heat-capacity region. The reduction of the neutron fraction above 10 keV in Row 3 is due to the reduced loading of U^{235} in this row.

The pulse characteristics of the two-region cores were evaluated using Monte Carlo techniques for the neutron generation time, ℓ , and using temperature-dependent cross sections in 2D S_N calculations to determine the reactivity loss with temperature. Point kinetics analysis was then used to estimate the corresponding minimum period. These results were confirmed using a space-time dependent, one-dimensional kinetics code with heat transfer.

A summary of results obtained for kinetic parameters for representative core configurations is given in Table IV. The greatest part of the negative temperature coefficient in the higher enrichment core designs is derived from the hydride region. The large negative temperature coefficient in the hydride region is due to both Doppler broadening and a strong spectrum hardening caused by the bound hydrogen scattering. The magnitude of this spectrum shift in the hydride region is shown in Figure 8. The thermal energy group is effectively upscattered to higher energies, causing a large negative reactivity effect. The minimum period attainable is slightly shorter with the BeO-UO₂ fuel due to a softer overall spectrum and the larger cell temperature effect of the Be moderator. The neutron generation time for both core designs is approximately the same due to the influence of the outer hydride region. The reactivity insertions required for the periods indicated in Table IV are in the range of \$3.16 to \$4.06. These values are only slightly smaller than that required for the current ACPR, which is \$4.40.

The pulse fluence improvement as a function of maximum HHC fuel temperature is shown in Figure 9. A maximum temperature rise in the hydride fuel of 1000°C was used in all calculations. These performance figures at selected temperatures are combined with the minimum period results in Figure 10. The two-region core concept requires a unique core design for a given maximum operating temperature. The actual calculations were performed at 1200°C for the UO₂-BeO and 2000°C for the UC-ZrC-C fuel. The results at 1400°C in the UO₂-BeO and 2400°C in the UC-ZrC-C are correct in performance improvement but are estimates for the minimum period. It can be seen that, in order to meet the performance improvement goal of 2.5, maximum adiabatic temperatures of about 1400°C will be required in the UO₂-BeO fuel. The minimum period at this temperature is approximately 2 ms. Although no specific minimum

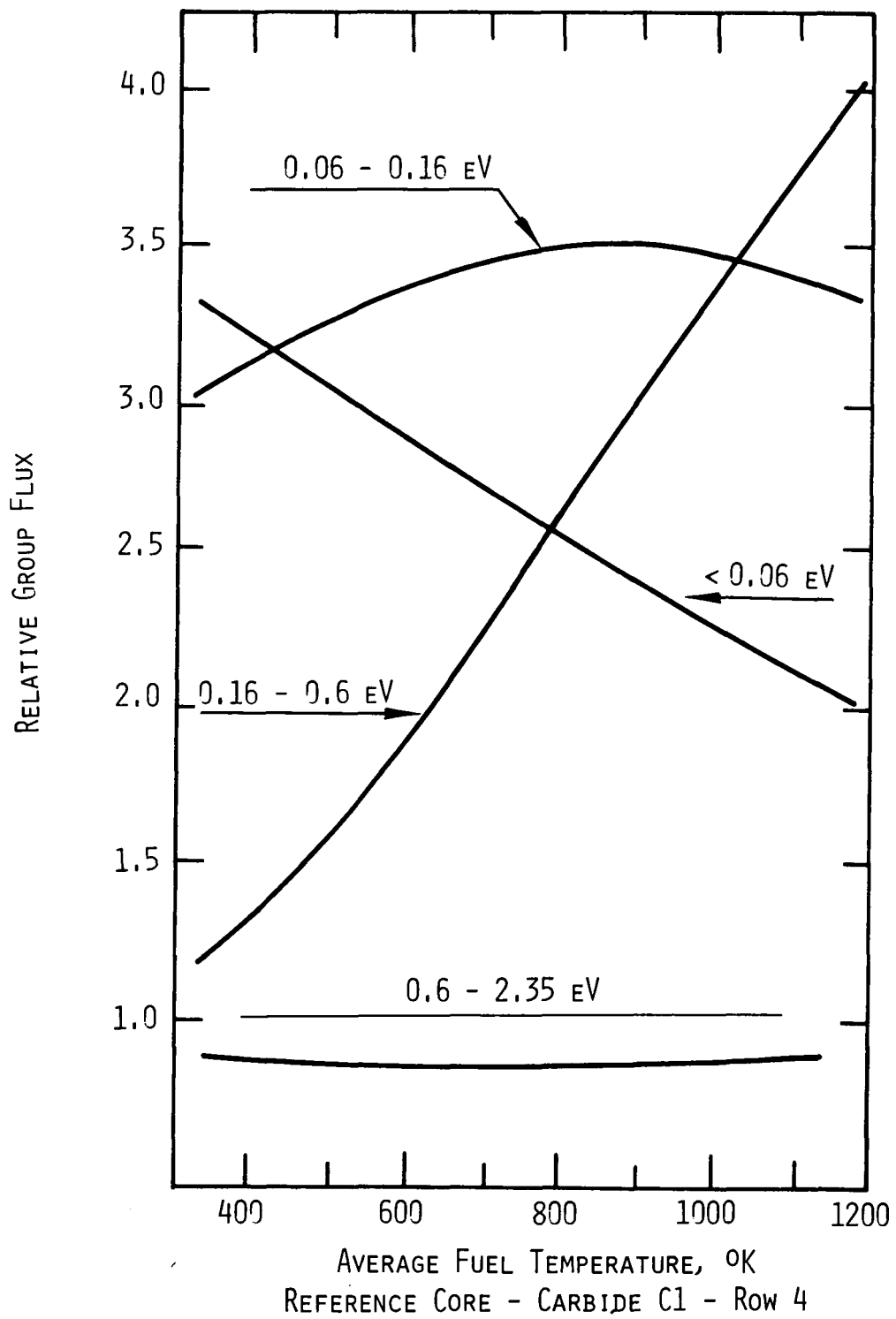


Figure 8. Change in Neutron Energy Distribution with Temperature in Hydride Region

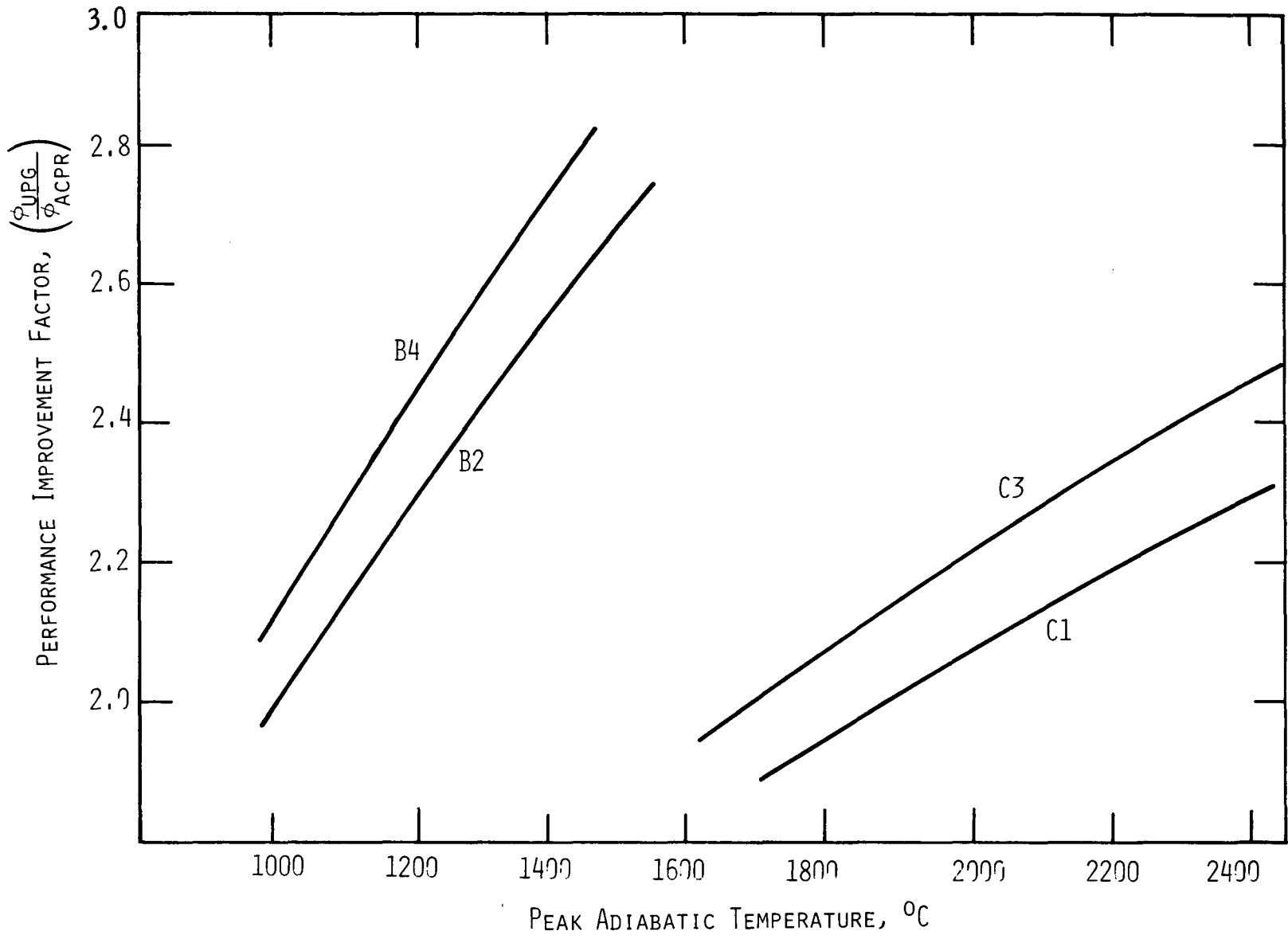


Figure 9. Pulse Fluence Improvement for $1000^{\circ}\text{C } \Delta T$ in Hydride ($900^{\circ}\text{C } \Delta T$ in ACPR)

TABLE IV

Kinetic Parameter Comparison

Core	Negative Temperature Coefficient (27°-500°C) ($\phi/^\circ\text{C}$)*	Neutron Generation Time (μs)**	Minimum Period (ms)	Performance Improvement Factor in HHC Fuel (max T °C)
ACPR	1.24	33	1.3	--
B2	0.88	38	1.7	2.3 (1200)
B4	0.77	43	2.3	2.4 (1200)
C1	0.85	37	1.8	2.1 (2000)
C3	0.67	41	2.6	2.3 (2000)
11 w/o Hydride Cell†	1.32			
ACPR Cell†	0.67			
BeO Cell† (B2)	0.26			
Carbide Cell† (C1)	0.15			

* $2D - S_N$ in $\phi/^\circ\text{C}$ in Hydride Fuel Region.

** Monte Carlo.

† 123 Group ID - S_N .

period was defined as a goal for the Upgrade, 2 ms is considered to be acceptable for the experiment program. The transient rod reactivity requirement to produce a 2 ms pulse for this core configuration is $\beta 3.60$. The control rod bank worth was calculated by Monte Carlo to be $\beta 9.20 \pm 0.80$. The steady-state flux at a nominal reactor power of 2 MW was 3.1×10^{13} n/cm² per second.

It is apparent from these results that the UC-ZrC-C fueled two-region core will not meet performance goals at reasonable operating temperatures. The primary reason for the reduced performance is the lower heat capacity of the UC-ZrC-C. The UC-ZrC-C fuel also requires more fissile material for a given reactivity worth, thereby increasing fission density peaking effects. The graphite-based fuels have obvious advantages in thermal shock resistance and in ease of fabrication. It was concluded that the superior performance potential and reduced temperatures in the UO₂-BeO outweighed the advantages of the graphite-based fuels, and the UO₂-BeO fuel was selected as the high-heat-capacity fuel for the ACPR Upgrade.

The range of design parameters surveyed in these studies was extended to configurations that had more reactivity worth in the hydride region than the reference cores, and also was extended to designs that had relatively small reactivity worth in the hydride region. The hydride-dependent cores provided large temperature coefficients and short pulse widths, but with somewhat lower pulse fluence. Core designs which depended heavily on the HHC fuel provided higher performance at the expense of longer periods. This trend is evident in the data of Figure 10. In general, core designs that had small reactivity worth in the hydride region provided no significant advantage over an all UO₂-BeO fueled design. The design and operational simplicity of the single-region reactor approach is a significant advantage. The development programs for both the UO₂-BeO and the UC-ZrC-C fuels have been successful in producing fuels that can be used for pulse applications. Due to the neutronic and heat transfer advantages of the UO₂-BeO, this fuel was used in a series of single-region core designs to evaluate this design alternative.

Single-Region Core Design Studies

1. U-ZrH_{1.5} Single-Region Core

Core designs utilizing only hydride fuel were performed early in the ACPR Upgrade project. Although relatively small pulse widths could be obtained, the

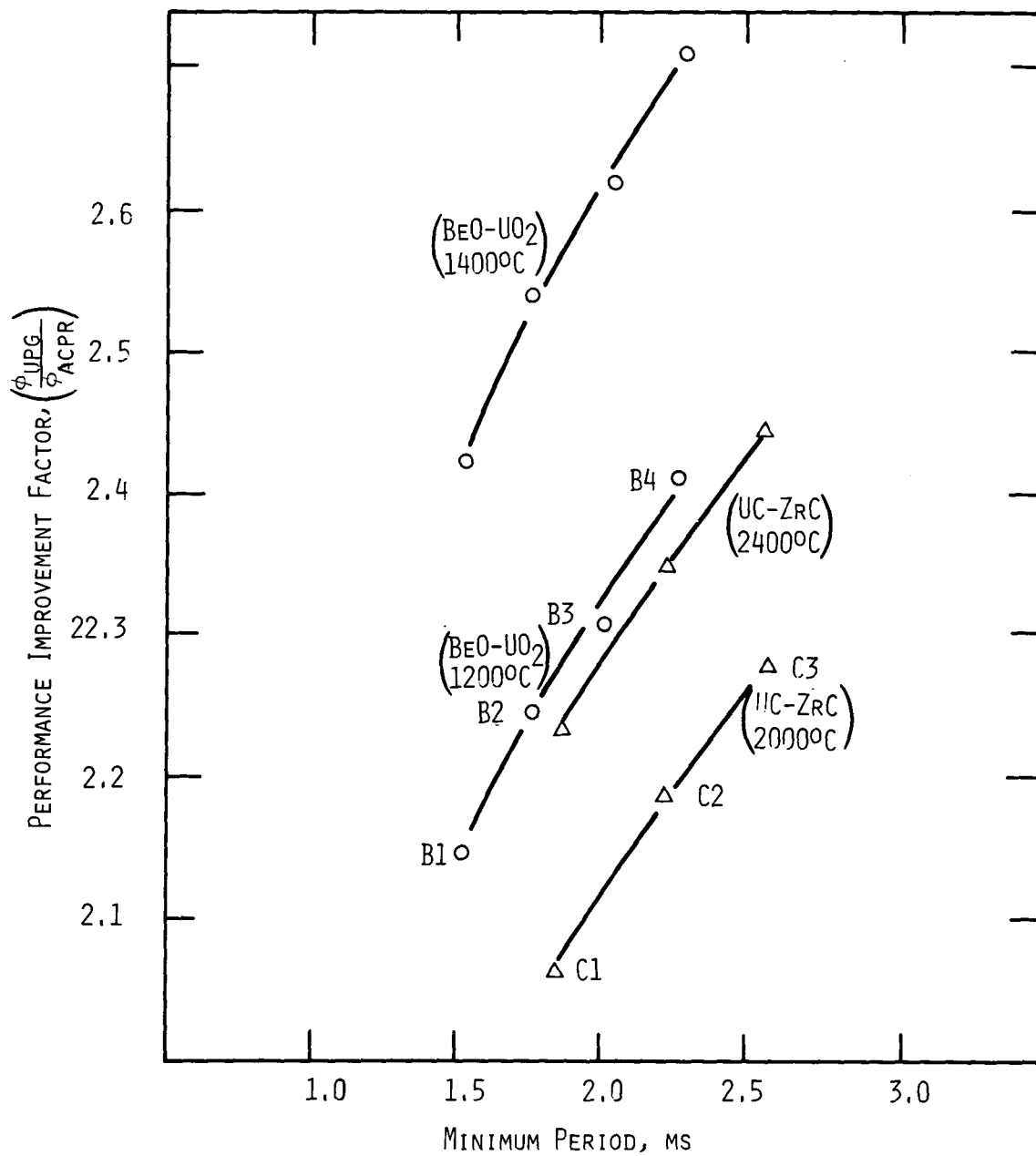


Figure 10. Performance/Period Comparison for Two-Region Core Designs

temperature limitations on the hydride fuel restricted performance improvement to values less than 1.5 at 1000°C. Since this was well below the performance goal of 2.6, no additional work was performed on this approach.

2. UO₂-BeO Single-Region Core

Following the selection of the UO₂-BeO fuel as the HHC fuel, a set of calculations was performed using UO₂-BeO in a single-region core. The elimination of the hydride (UO₂-BeO) interface allows a higher average fission density in the core for a given peak temperature and results in higher pulse fluence. The smaller negative temperature coefficient results in larger pulse widths for this core configuration. The results of these studies showed that the neutron generation time could be reduced significantly by utilizing nonhydrogenous reflector elements at the outer edge of the core. The pulse widths attainable are comparable to those in the hydride driven cores for equal performance levels. To obtain smaller pulse widths in the hydride cores requires some performance degradations.

The survey calculations and preliminary designs studies will be covered in detail in the next quarterly progress report. A brief summary of important design considerations and a comparison of a reference single-region core design will be presented in this quarterly report.

The preliminary reference single-region core design which was calculated to provide a comparison between the single- and multiple-region core designs is described in Table V.

All fuel elements in this configuration are of uniform loading. The performance and kinetic characteristics of this core are shown in Table VI. At 1200°C maximum fuel temperature in the UO₂-BeO, there is a performance advantage with the single-region core with only a modest period penalty. The reactivity required for pulsing is significantly reduced, which indicates a lower worth transient bank requirement. Steady-state capability is not significantly different for the two designs. There are two additional advantages to the single-region core indicated in Table VI. The reactivity worth of the outer elements will be significantly higher in the single-region core, allowing considerably more flexibility in reactivity adjustment for negative experiments or initial core loading. Early results indicate

TABLE V

Single-Region UO₂-BeO Core Configuration

Number of Fuel Elements	200
Reflector Elements	55-304 Clad Nickel Elements
w/o U ²³⁵ O ₂	7.0
w/o U ²³⁸ O ₂	14.0
Control Rod Bank	6 Fuel Followed
Transient Rod Bank	3 Void Followed
Safety Rod Bank	1 or 2 Fuel Followed

TABLE VI

Comparison of One- and Two-Region
 UO_2BeO Cores at 1200°C or 1400°C*

<u>Characteristics</u>	<u>Two-Region Core</u>	<u>One-Region Core (UO_2BeO)</u>
Temperature Coefficient ($\phi/^\circ C$)	0.80**	0.48
Neutron Generation Time (μs)	40	24
Minimum Period (ms)	2.0 (1.9)	2.2 (1.9)
Performance Improvement Factor (1200/1400°C)	2.35 (2.60)	2.60 (3.1)
Pulse Reactivity for 1200°C	3.60	2.50
<u>Steady-State Flux</u> Power Density $\left(\frac{n/cm^2 \cdot second}{W/cc}\right)$	1.45×10^{12}	1.60×10^{12}
Fuel Element Worth (ϕ):		
5th Row	-	22.6
6th Row	9.2	13.7
7th Row	5.2	-
Fuel Pellet P/C	1.5 - 1.9	1.35 - 1.6

* Values for 1400°C shown in parentheses.

** Per °C in $U-ZrH_{1.5}$.

about a 10% higher worth for fuel motion lots or moderated experiments in the single-region core. The number of fuel elements required to make up this reactivity is, however, considerably smaller in the single-region core. The stress in the individual fuel pellets is also reduced in the single-region core. The P/C values shown in Table VI indicate a reduction of about 25 percent in the peak stress for the single-region core.

The comparison of the one- and two-region UO_2 -BeO core designs reveals several significant advantages for the all BeO-based core. The performance advantage of the all-BeO core is easily anticipated due to the elimination of the two-region interface. The reduction in neutron generation time that allows a favorable pulse width to be obtained from this approach is not as obvious, and the series of calculations leading to this result will be covered in the next quarterly report. The combination of single-region core advantages and the uncertainty of the new $\text{U-ZrH}_{1.5}$ fuel behavior led to a design decision to select the all-BeO single-region approach for the ACPR Upgrade.

The technical advantages of the all- UO_2 -BeO core, not the least of which is an overall design and operational simplification, would suggest that this approach would have merited further consideration earlier in the design studies. The primary reasons why this design decision was delayed until the final phase of these studies are the following:

1. Early UO_2 -BeO testing was not encouraging until the development of the cold-pressed and sintered material. This material also could be produced within cost constraints and in reasonable time frames.
2. The UC-ZrC-C single-region core designs were not as favorable as the UO_2 -BeO from the viewpoint of heat transfer, performance, and kinetic characteristics. The fuel selection decision was an important prerequisite to the single-region core design decision.
3. The somewhat smaller pulse widths obtainable with the hydride cores have taken a lower priority than performance potential, making the single-region core more attractive.

4. Hydride fuel performance did not meet expectations. If hydride fuels could have been operated at the 1100-1200°C range as originally anticipated, the two-region core concept would provide more performance with shorter periods for a given UO_2 -BeO temperature. Early tests suggest the hydride fuel will be limited to $\leq 1000^\circ\text{C}$, in its current form. At this hydride temperature, a single-region UO_2 -BeO core is preferable.

The survey calculations for the single-region UO_2 -BeO core have been completed, and a series of final design calculations has been initiated. These calculations are focused on a nominal 200-element core with 33% enriched UO_2 and nickel reflector elements. The details of these calculations and final neutronic design results will be presented in the next quarterly report.

CHAPTER III

TASK 3. ACPR UPGRADE CONSOLE DEVELOPMENT W. H. Sullivan, 5423

Negotiations on the console price, delivery, and technical details were completed, and the contract was placed on October 15, 1976. The final price was within the original budget estimate, and the quoted delivery of 12 months will meet the required time schedule.

Detailed design of the linear and log power monitoring channels is complete. The drawings have been reviewed and approved and are now being drafted in final form, prior to the start of fabrication.

Detailed design of the plant protection system instrumentation and power supply drawers is approximately 30 percent complete.

CHAPTER IV

TASK 4. MECHANICAL DESIGN

H. E. Walling, 1136

Introduction

This task involves the design and installation of the cooling system, penthouse and hoist, and reactor tank modifications.

Cooling System

Plant Engineering submitted its final version of the cooling system. A purchase requisition was approved by Laboratory Management for procurement and installation of the final version.

Penthouse Addition

Plant Engineering modified their original penthouse design to make provision for a larger addition. The larger addition will require more structural stiffening to the present ACPR high bay.

Ventilation System

Conceptual design for the cavity ventilation system was finished and detailed design begun. Plant Engineering will complete the detailed design next quarter.

Experiment Loading Facility

A detailed design of the experiment loading facility was begun. Requisitioning for this apparatus will begin next quarter. Manufacture and delivery should be complete next quarter.

Coolant Inlet Manifold

A conceptual design for modification to the ACPR tank liner was begun. The purpose of the modification is to insure that cold water will be available below the core region. Modification contains dampers to control flow to the core.

Bridge Support Structure

A conceptual design for a new bridge to replace the present cantilevered structure was begun.

Fuel Motion Shield

Negotiations were begun with Department 5420 to minimize the interference between the radiography facility and the fuel motion facility in the ACPR tank. Conceptual design will start next quarter.

CHAPTER V

TASK 5. FUEL ELEMENT DESIGN

J. A. Reuscher, 5424; F. M. Morris, 5424;
H. E. Walling, 1136

Introduction

This task involves the design, testing, and procurement of the high-heat-capacity fuel elements for the central region of the core. The results of the in-pile fuel experiments are used to develop the fuel material and the fuel region configuration. The major activities for this quarter included the decision for the high-heat-capacity fuel and the element section tests for both BeO-UO₂ and (UC-ZrC)-graphite to the energy depositions required for the Upgrade core. In addition, final heat transfer calculations for the BeO-UO₂ fuel-element design were conducted.

Element Section Test

The element section test apparatus was described in the previous quarterly report⁸ along with the initial test results for the (UC-ZrC)-graphite fuel. During this quarter, the BeO-UO₂ element section test was conducted to a peak adiabatic fuel temperature rise of 1180°C; and the (UC-ZrC)-graphite tests were extended to a maximum temperature rise of 2300°C. These experiments included fuel and clad thermocouples; fission product measurements were conducted to determine the average energy deposition and the energy deposition profile. The BeO-UO₂ tests included the niobium cups as inner liner, and the (UC-ZrC)-graphite tests included the Carbitex inner liner. The results of the element section tests are given in Figure 11; maximum fuel energy deposition is shown as a function of the clad temperature rise for both fuels. The (UC-ZrC)-graphite results are shown for U-235 densities of 400 and 600 mg/cc; there is one data point without water around the clad.

The clad temperature rises shown in Figure 11 are quite modest. For a peak energy deposition of 500 cal/g, the clad temperature rise is about 97°C for BeO-UO₂ and 120°C for (UC-ZrC)-graphite. The clad temperature rise is given in Figure 12 as a function of maximum adiabatic fuel temperature rise. This figure shows that the clad temperature rise depends strongly on the fuel temperature rise rather than on

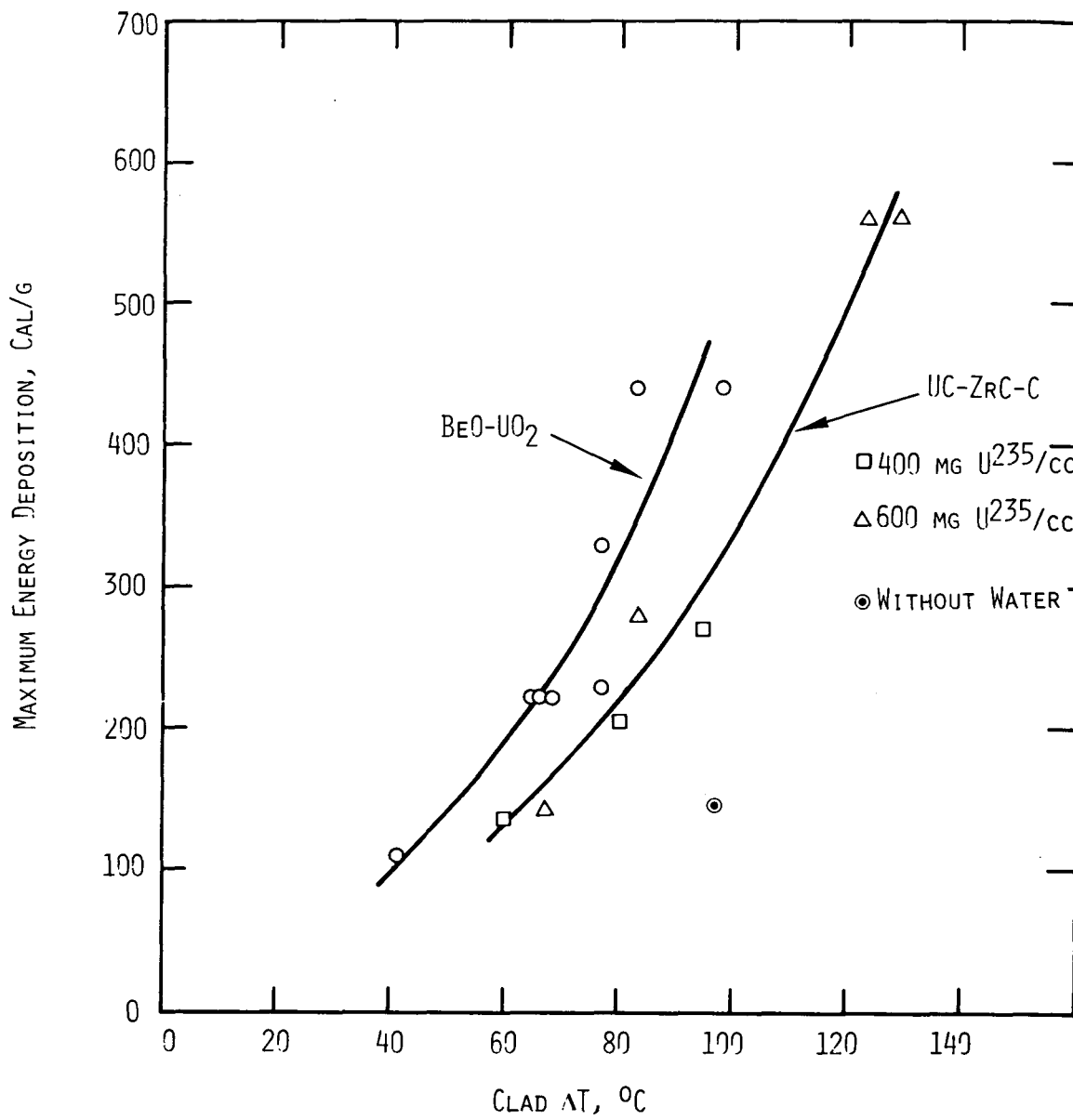


Figure 11. Element Section Test

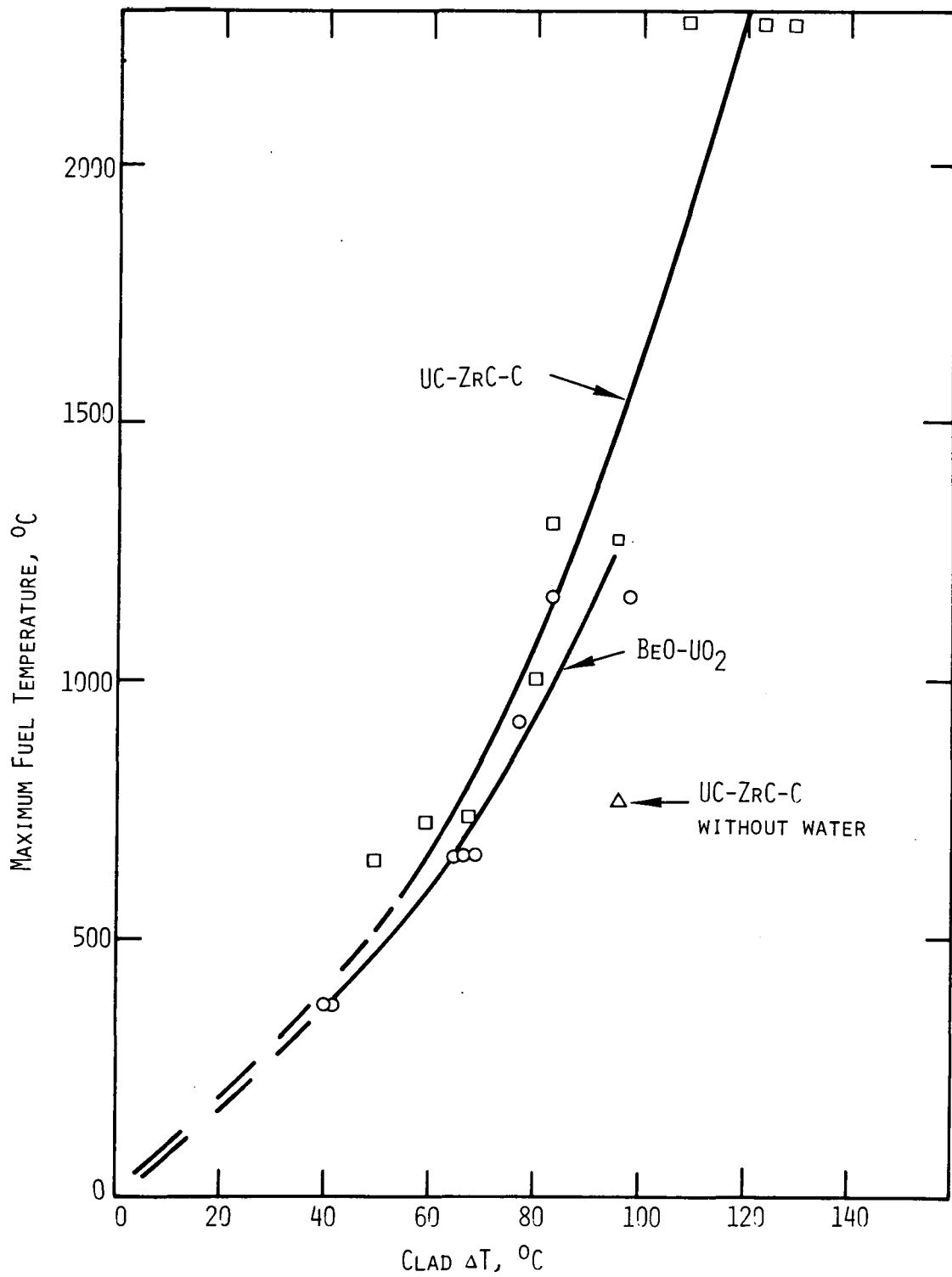


Figure 12. Element Section Test

the energy deposition. However, for the peak energy deposition required for the Upgrade core (~500 g/cc), the BeO-UO₂ fuel-element design produces the lower clad temperature.

High-Heat-Capacity Fuel Decision

On October 19 and 20, 1976, the fuels development programs for the ACPR Upgrade were reviewed by a fuel selection committee. This committee consisted of the following individuals:

J. A. Reuscher, 5424, Sandia Laboratories
 P. S. Pickard, 5424, Sandia Laboratories
 T. R. Schmidt, 5421, Sandia Laboratories
 W. B. Shepard, 9512, Sandia Laboratories
 S. O. Johnson, Intermountain Technologies, Inc.
 W. K. Barney, Argonne National Laboratory

The (UC-ZrC)-graphite and BeO-UO₂ fuel development programs were reviewed in detail with emphasis being placed on the in-pile test results. The core physics aspects of both fuels were discussed; the fuel comparison was based primarily on the fuel temperature rise required for a certain performance improvement factor. The core physics data are summarized in Table VII.

TABLE VII

Summary of Core Physics Data

<u>Core</u>	<u>Max Temperature Rise, °C</u>	<u>Performance Improvement</u>
UC-ZrC-C	2000	2.2
UC-ZrC-C	2400	2.35
BeO-UO ₂	1200	2.3
BeO-UO ₂	1400	2.6

The cost and schedule considerations were reviewed for each fuel. The heat transfer calculations and element section test results were examined, as well as the safety analysis considerations for each fuel.

After review of the fuels data, the committee recommended that BeO-UO₂ be chosen as the high-heat-capacity fuel for the ACPR Upgrade for the following reasons:

1. The BeO-UO₂ offers a greater performance improvement with less fuel temperature and lower clad heat flux than the (UC-ZrC)-C.
2. The BeO-UO₂ offers a core with a greater performance improvement at a shorter initial period than the (UC-ZrC)-C.
3. There is a greater potential for extending the core performance with the BeO-UO₂.
4. The BeO-UO₂ cost is higher than the (UC-ZrC)-C; however, the cost is within the budget for the central region fuel.

BeO-UO₂ Fuel Element Heat Transfer Calculations

During this quarter, refined heat transfer calculations were conducted for the BeO-UO₂ fuel element design. These computations were intended to refine the previous calculations with particular attention given to the "hot spot" caused by the horizontal niobium layer between niobium cups.

The model for the calculations is shown in Figure 13. In the actual fuel element there will be from 5 to 10 cups; however, because of limitations in the computer code TAC-2D,¹⁰ only two cups were analyzed. The so-called "hot spot" is that vertical section of the liner and clad near the bottom of the upper niobium cup.

Transient and steady-state calculations were performed with various fuel-liner and liner-clad gap thicknesses. The results of the transient calculations are given in Figures 14 and 15. The clad heat flux is shown as a function of axial distance along the fuel element. Figure 14 gives the maximum heat flux which occurs at 4 seconds for a gap of 0.025 cm (0.010 inch) between the liner and clad with helium in the gap. The niobium cup creates a heat transfer path to the clad as is shown by the peaking in the heat flux at the location of the cup bottom. The peak fuel energy deposition is 630 cal/g, which corresponds to a maximum adiabatic temperature

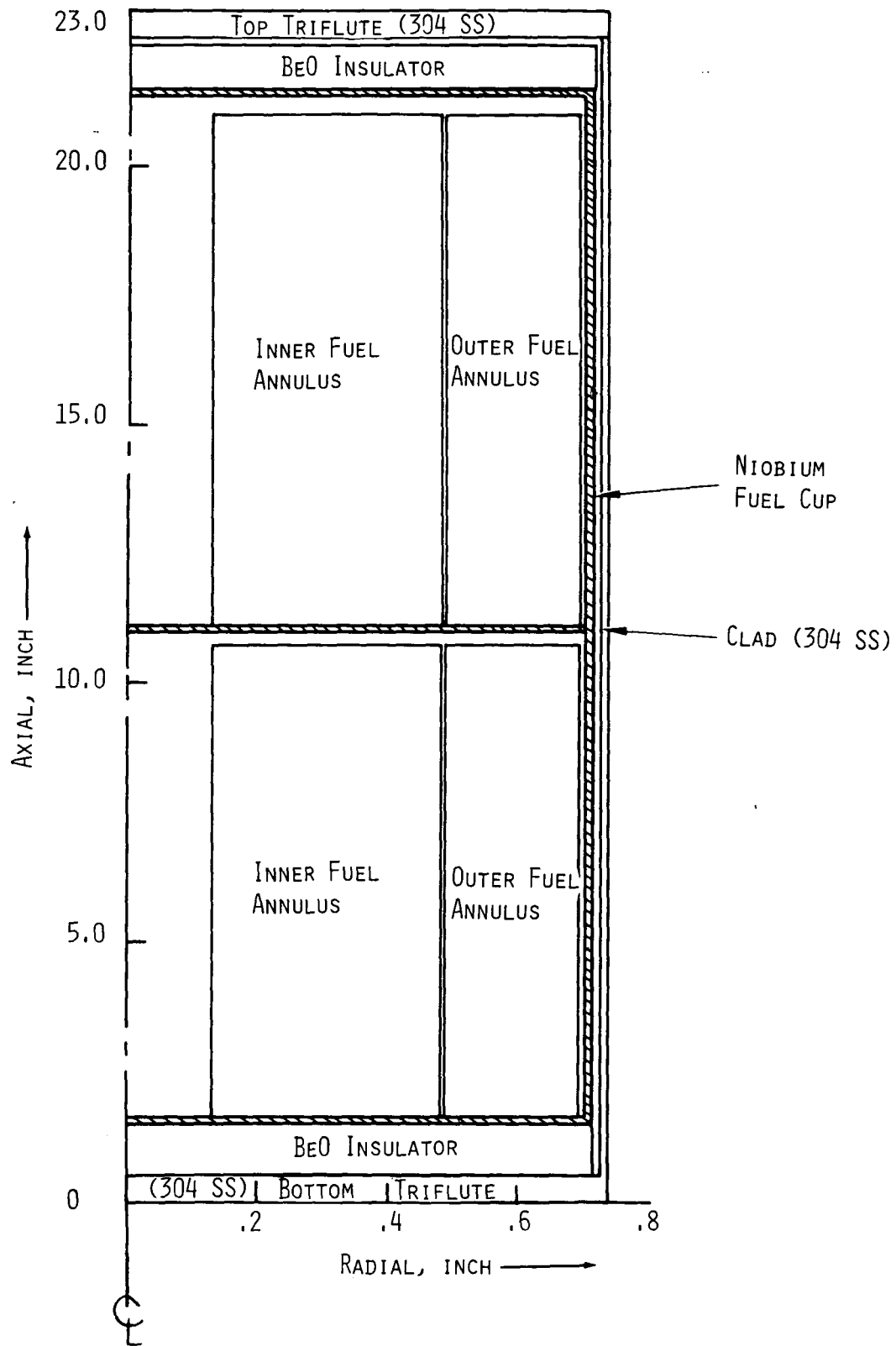


Figure 13. BeO-UO₂ Fuel Element Design

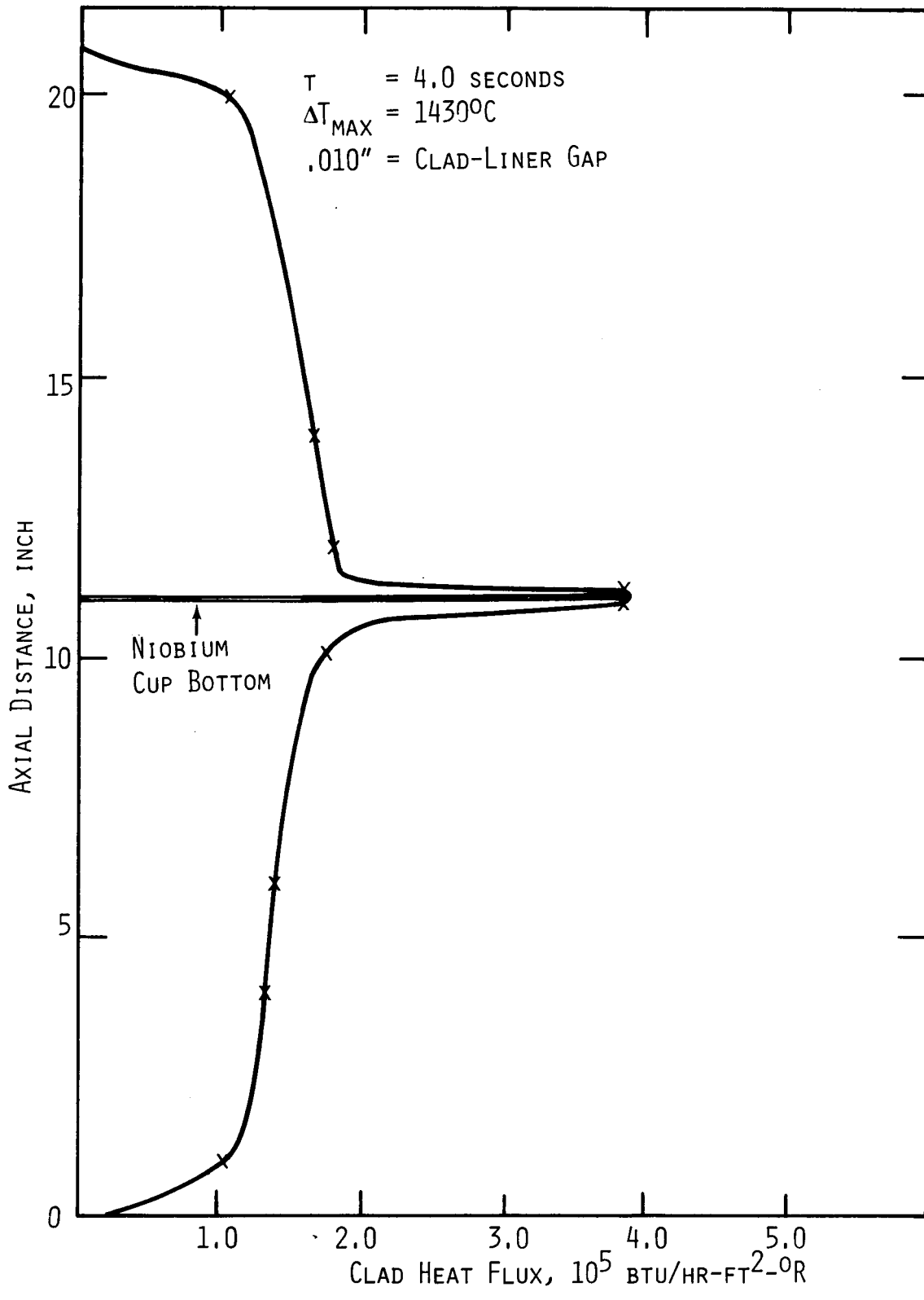


Figure 14. Peak Clad Heat Flux versus Axial Distance

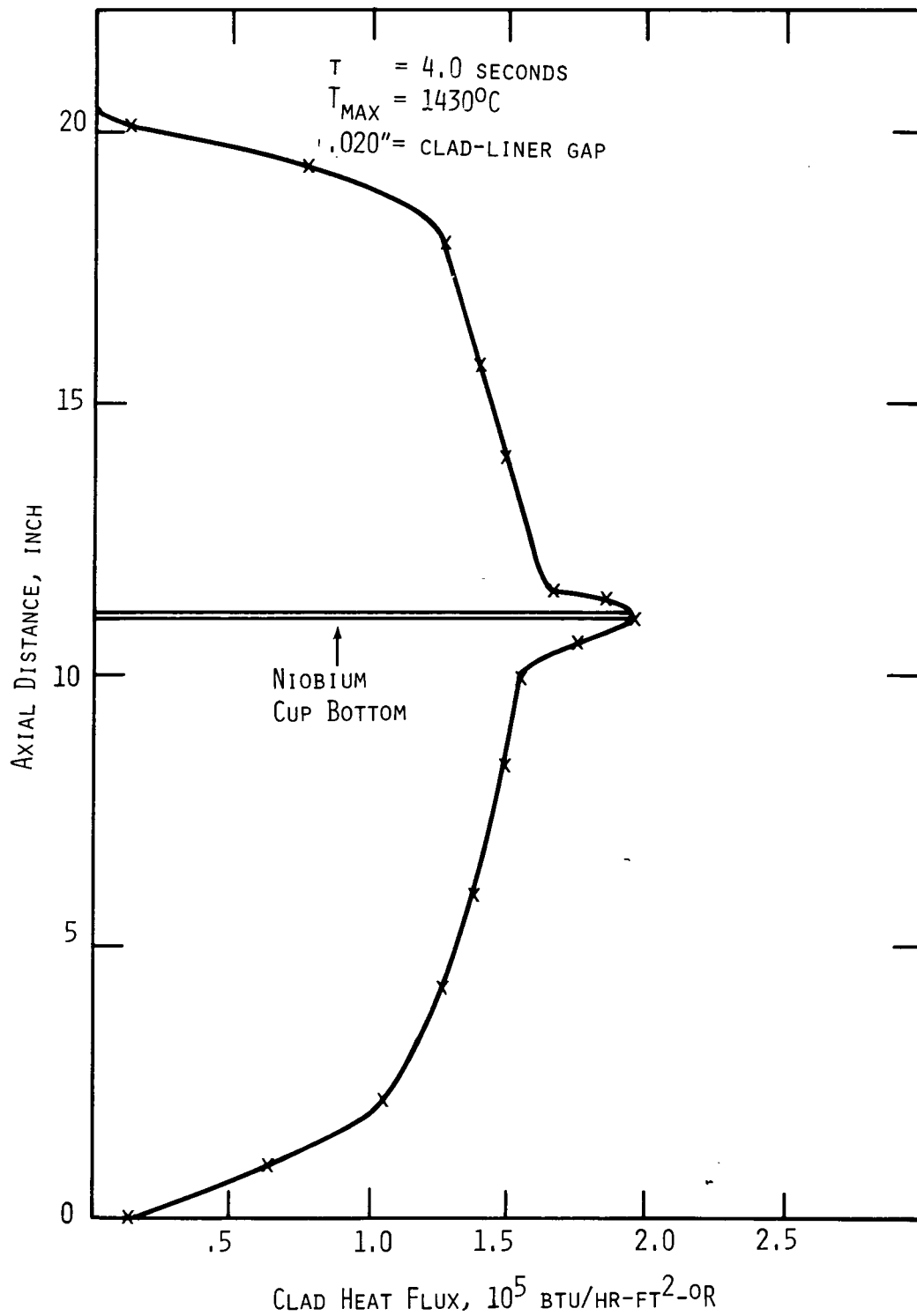


Figure 15. Peak Clad Heat Flux versus Axial Distance

rise of 1430°C. The "hot-spot" heat flux exceeds the Rohsenow pool boiling critical heat flux of 3.5×10^5 Btu/hr-ft²; however, there is only a small increase in the clad surface temperature. The same calculation is shown in Figure 15 for a liner-clad gap of 0.051 cm (0.020 inch). With the larger liner-clad gap, the heat flux peaking at the cup bottom is reduced by a factor of two and is well below the Rohsenow correlation in magnitude.

The results of steady-state calculations are given in Figures 16 and 17. The clad heat flux at the niobium cup bottom is shown as a function of average element power for two fuel-liner and liner-clad gaps with helium fill gas. These results indicate that the departure from nucleate boiling heat flux will not be exceeded at the design steady-state power (~ 10 kW per element). The maximum fuel temperature as a function of average element power is shown in Figure 17 for two-gap configurations. These results indicate that the fuel temperature at the design steady-state power will not be excessive.

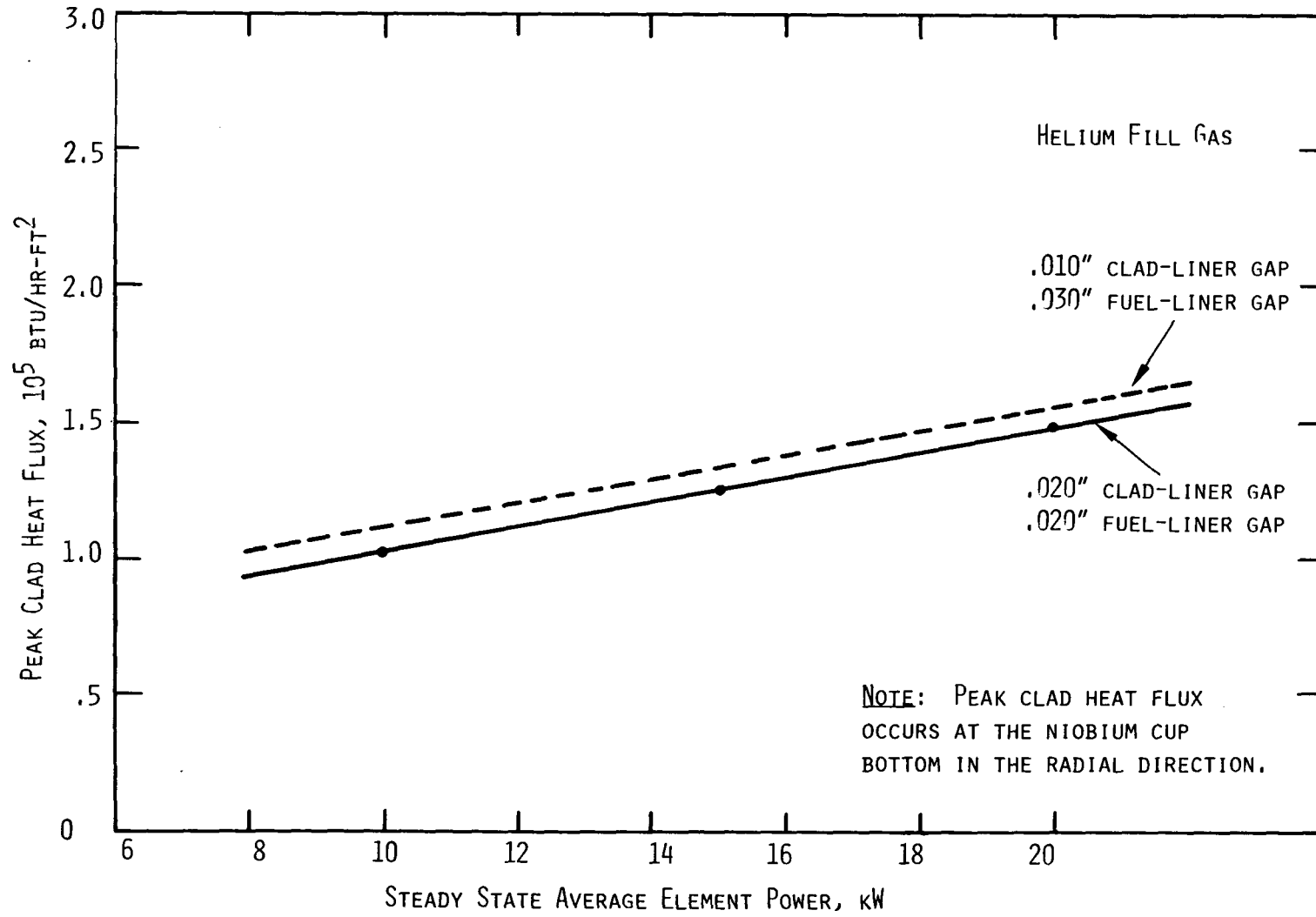


Figure 16. Heat Flux versus Element Power

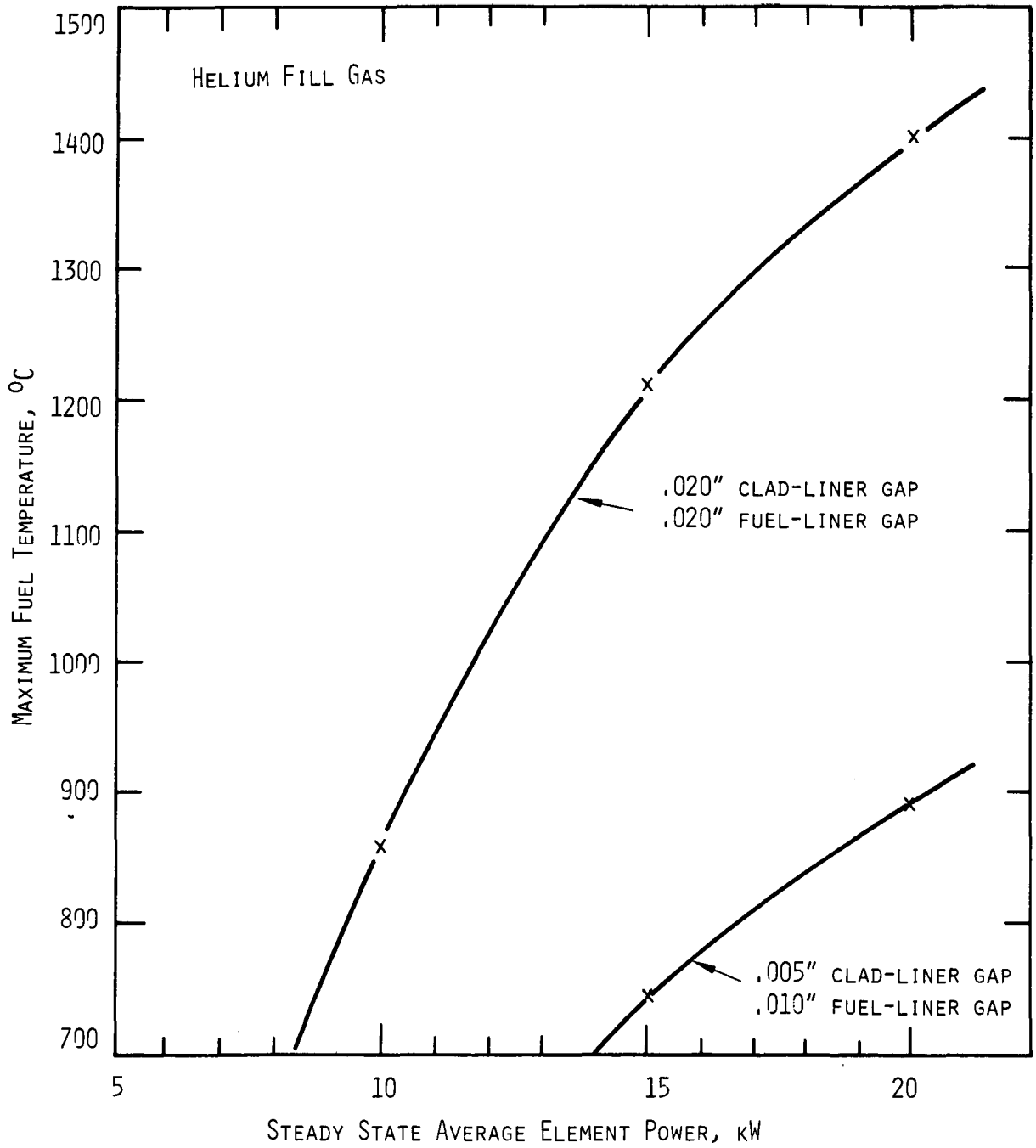


Figure 17. Maximum Fuel Temperature versus Element Power

CHAPTER VI

TASK 6. BeO-UO₂ FUEL DEVELOPMENT D. J. Sasmor, 5422⁷; P. S. Pickard, 5424

Single pulse testing of the BeO-UO₂-ZrO₂ dual annuli fuel samples has been completed. Two experimental containers, one with 0.5-inch and one with 0.75-inch polyethylene moderator, have been exposed to 50, 80, and 100 percent of maximum reactor pulses. Results of the visual examination which followed each of the individual pulses are given in Table VIII. The calculated outer annuli temperatures are based on fission profiles from monitor pellets exposed at the 50-percent level. One fracture was observed at the 100-percent pulse, 0.75-inch polyethylene. The results of the single pulse test of the fuel with ZrO₂ parallel to that of the tests of the fuel without ZrO₂.

Both test vehicles and fuel columns have been reassembled for multipulse testing at maximum reactor pulse. The column with 0.75-inch polyethylene moderator has been exposed to 50 pulses (completed Dec. 10) but has not been examined as yet.

Two additional fuel-element section tests were completed this past quarter. The first was a buildup series consisting of four pulses which were 25, 50, 75, and 100 percent of maximum reactor pulse; the second was a single pulse test at 100 percent of maximum reactor pulse. In the buildup series test, there was approximately a 1-hour cool-down time between pulses; therefore, starting conditions varied for each pulse. Table IX is a summary of all results of the BeO-UO₂ fuel element section test to date.

The compatibility studies, BeO-UO₂/Nb, were contained this past quarter using the hot-pressed BeO-UO₂ containing 0.005-inch diameter Nb wire. The test samples were placed between two niobium plates, placed in a tungsten crucible, and heated for 4 hours at temperature in flowing tank helium. Contact pressure between the samples and the niobium plates was due only to the weight of the samples and plate. "In toto", five samples were used; one sample was exposed to 4 hours at 1773 K (1500°C) and 4 hours at 1973 K (1700°C), while two were exposed to 1773 K and 1973 K,

TABLE VIII

BeO-UO₂-ZrO₂ Fuel Test Cold-Pressed and Sintered
0.25-Inch High Dual Slotted Outer Annuli

<u>Percent Pulse</u>	<u>Moderator Thickness (in.)</u>	<u>Outer Annuli</u>		<u>Number Samples*</u>	<u>Number Fractures*</u>
		<u>Average Temperature (K)</u>	<u>Maximum Temperature (K)</u>		
50	0.5	810	883	10	0
80	0.5	1067	1179	10	0
100	0.5	1231	1367	10	0
50	0.75	889	1011	10	0
80	0.75	1188	1371	10	0
100	0.75	1378	1598	10	1

* Exclusive of end pairs.

TABLE IX

Fuel Elements Section Tests

Pulse	Moderator (in.)	Outer Annulus Temperature (°C)		ΔT Niobium Clad Region (K)	ΔT Bulk Water Region (K)	ΔP Fuel Region (psi)	ΔP Water Region (psi)
		Average	Peak				
50	0.5	769	928	49	49	14	3.5
50	0.75	835	949	77	58	16	4
25*	0.5	550	640	41	16	7	1.4
50*	0.5	768	927	68	39	13	5
75*	0.5	966	1188	77	64	18	15
100*	0.5	1153	1433	83	103	24	38
100	0.5	1153	1433	100	67	9	12

* Build-up series.

respectively. Except for one sample at 1773 K where there was niobium- (plate) to-niobium (wire in sample) cohesion, none of the samples exhibited any sticking or adhesion. The niobium plates appear to have undergone considerable grain growth and etching on those surfaces exposed to the BeO-UO₂. The BeO-UO₂ samples appear to have undergone some erosion of the BeO, leaving the UO₂ + Nb wires in relief. This was most evident on the samples exposed at the highest temperature. The samples are undergoing optical and scanning microscopic examination to determine the extent of reaction. The niobium plates will be examined by Auger spectroscopic techniques to determine if the BeO has transferred to the niobium plates. The furnace parts were "swiped" and checked for Be; the reports show negligible quantities on the crucible and furnace parts.

CHAPTER VII

TASK 7. GRAPHITE FUEL DEVELOPMENT C. H. Karnes, 5847; R. H. Marion, 5847

The development of the (U,Zr)C/graphite fuel is nearing completion with two additional lots of fuel being received during the reporting period.

Lot SLO22 is a fuel containing 0.626 g/cc of fully enriched uranium. This fuel was fabricated with a high uranium loading in order to have pellets which would reach significant temperatures in a fuel-element section test. Only reactor pellets were fabricated from this lot of material because the fuel-element section tests were primarily intended to verify the heat transfer calculations. These results are presented in another section of this report. Even though this material contained 24.8 volume percent of carbide phase, which is too high to have good thermal stress resistance properties, one of the 1-inch-high solid pellets in the section test survived a peak temperature of 2330°C with a peak-to-minimum energy density ratio of 3.02. The other pellets in the test were either-end pellets or had thermocouple holes drilled into them, resulting in fractures.

In order to assess the fabrication techniques and properties of a fuel material having the maximum uranium content which would be considered for the upgraded ACP core, Lot SLO24 was produced at LASL which contained 0.8 g/cc of 62 percent enriched uranium. The enrichment was chosen to allow significant temperatures and stresses to be reached in the current reactor. Lot SLO24 has been received but the pellet tests have not yet been performed.

Mechanical properties were determined on Lot SLO24 at temperatures up to 2400°C. These properties are shown in Table X.

TABLE X

Mechanical Properties of Lot SL024

<u>Sample No.</u>	<u>Test Temp (°C)</u>	<u>Fracture Stress (psi)</u>	<u>Fracture Strain (%)</u>	<u>Initial Modulus (10⁶ psi)</u>
SL024-0221-1	22	2990	0.26	1.7
SL024-0221-2	1005	3460	0.28	1.2
SL024-0226-1	1000	2520	0.24	1.1
SL024-0228-3	1475	2460	0.25	1.2
SL-24-0231-3	1475	2570	----	----
SL024-0231-2	1970	2750	0.95	1.1
SL024-0226-3	2010	2650	0.57	0.8
SL024-0224-1	2390	990	0.67	0.31
SL024-0228-1	2400	970	0.75	0.28

It was previously reported that Lot SL007, a hot-pressed solid pellet material, survived pulse tests that produced peak temperatures of 2000°C with a peak-to-minimum ratio of 2.2. Attempts to further improve Lot SL007, which contained 30 volume percent carbide phase, have been made by fabricating Lot SL021 with 17 volume percent carbide. The material has not been characterized completely, but initial indications are that it may be superior to the other hot-pressed materials. A compilation of mechanical properties obtained are shown in Table XI. Unlike most of these materials, Lot SL021 material did not exhibit a measurable creep rate at temperatures up to the maximum shown in Table XI.

There is a small amount of additional mechanical and thermal characterization work to be done on these materials. These studies along with the required reactor pulse tests will be performed in the near future in order to complete the documentation of the (U-Zr-C)-graphite fuel development program.

TABLE XI

Mechanical Properties of Lot SL021

<u>Sample No.</u>	<u>Test Temp (°C)</u>	<u>Fracture Stress (psi)</u>	<u>Fracture Strain (%)</u>	<u>Initial Modulus (10⁶ psi)</u>
SL021-0179-1	22	4580	0.39	2.1
SL021-0180-3	22	4420	0.51	1.9
SL021-0179-2	1005	4030	0.22	1.8
SL021-0184-1	1010	> 4770	> 0.22	2.2
SL021-0179-3	1490	> 2670	> 0.25	1.0
SL021-0184-2	1490	3060	0.55	0.9
SL021-0180-1	2000	> 3380	> 0.59	1.2
SL021-0184-3	2000	> 3790	> 0.6	1.1
SL021-0185-2	2000	4240	1.75	1.0
SL021-0180-2	2410	1120	0.62	0.3
SL021-0185-1	2390	1470	1.4	0.22

CHAPTER VIII

TASK 8. DRIVER CORE FUEL ELEMENT J. A. Reuscher, 5424; F. M. Morris, 5424

Introduction

The outer region of the Upgraded ACPR may consist of uranium-zirconium hydride fuel elements similar to the fuel elements in the present ACPR core. This task is responsible for the in-pile testing and design of the U-ZrH elements.

In-Pile Tests

A number of in-pile tests were conducted with the uranium-zirconium hydride specimens manufactured by General Atomic Company. All of the samples had a diameter of 3.56 cm and lengths from 0.32 to 12.7 cm. The experiments were conducted with a stainless-steel test apparatus similar to the one used with the element section tests. The U-ZrH irradiations were conducted without water around the clad vessel, and the internal pressure in the vessel was monitored as well as the fuel temperature.

The samples were cast with 6 weight percent of fully enriched uranium, and some of them were zone melted to densify the uranium-zirconium alloy. The samples were then hydrided and finish-machined to tolerance dimensions. Some of the samples were hydrided to the proper H-to-Zr ratio during the first pass in the hydride furnace; other samples required additional hydriding, and some were dehydrided to reduce the ratio.

A number of 2.5-cm-thick samples were tested in a 10-cm-long stack of pellets. The initial experiments produced peak temperatures near 1300°C, and numerous cracks were produced in the samples. In a qualitative sense, the zone-melted material showed more severe cracking than the as-cast material, and there were some small particles created by the fracturing.

Since the fuel would be manufactured in 12.7-cm lengths, another test was conducted with a piece of zone-melted fuel with an H-to-Zr ratio of 1.49. The experiment had the following energy deposition and deposition profile characteristics for a maximum ACPR pulse.

average energy deposition	-	120 cal/g
peak energy deposition	-	166 cal/g
average temperature rise	-	820°C
peak temperature rise	-	1020°C
peak to average profile	-	1.38
peak to minimum profile	-	1.92

This specimen was first hydrided to H/Zr = 1.42, then to H/Zr = 1.59 and, finally, it was dehydrided to H/Zr = 1.49.

The initial test series consisted of 10 full-size pulses. The temperature of the specimen was allowed to approach ambient prior to the next pulse. The pressure in the test vessel rapidly reached 120 psi within milliseconds after the pulse and slowly decayed within seconds to about one-half of the peak value. The pressure returned to its initial value as the temperature returned to ambient.

After the 10 pulses, the experiment was disassembled and visually examined. Several cracks were evident as shown in Figure 18. The 12.7-cm piece had broken into two pieces; there were longitudinal cracks in these pieces. The top of the experiment is to the left of the photograph; there was a 0.64-cm-thick pellet of fuel at the top and a 1.27-cm-thick pellet at the bottom of the large piece. The end pellets reduced temperature peaking at the ends of the 12.7-cm-long fuel pellet. Approximately 100 milligrams of powder were separated from the fuel pieces. The powder was very fine and small in size, although readily visible to the unaided eye.

The fuel pieces were reassembled in the stainless-steel container, and the experiment was again placed in the reactor and more pulses were produced. Ten 100%-of-maximum pulses were followed with 75%-, 80%-, 90%-, 95%-, and two 100%-of-maximum pulses. For the maximum pulses, the peak pressures reached about 160 psi, and the time behavior of the pressure was similar to that during the first 10 pulses. The experiment was again disassembled, examined visually, and photographed. The picture of the fuel specimens after 27 pulses is shown in Figure 19. The axial cracks are more evident and appeared to propagate with the additional pulses. More fine powder (~ 60 mg) was separated from the fuel specimens.

A photograph of the fracture surface of the lower half of the 12.7-cm-long specimen is shown in Figure 20. The fracture is very rough, indicating a large grain structure, and there is evidence of a thin "case" around the central core of the specimen. The exact nature of this "case" has not been determined.

Further tests are required before the U-ZrH can be proven satisfactory for use in the two-region ACPR Upgrade core. New test specimens will be obtained with a larger H/Zr of about 1.6, and the dehydriding process will not be used. There is also concern that the original U-ZrH production process used with the ACPR core can no longer be employed. The original process used in 1966 hot-extruded the uranium-zirconium alloy and then the fuel was hydrided. This production process is no longer available.

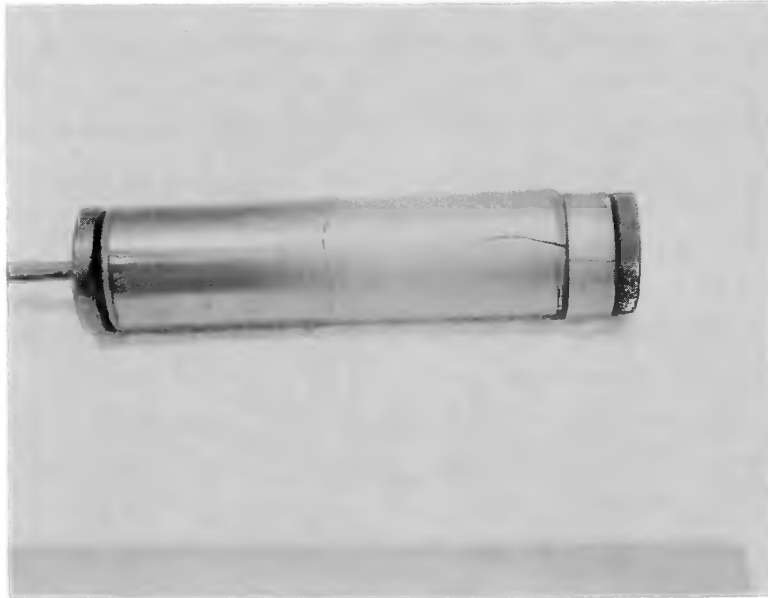


Figure 18. Photograph of $\text{U-ZrH}_{1.49}$ Specimen after 10 Pulses

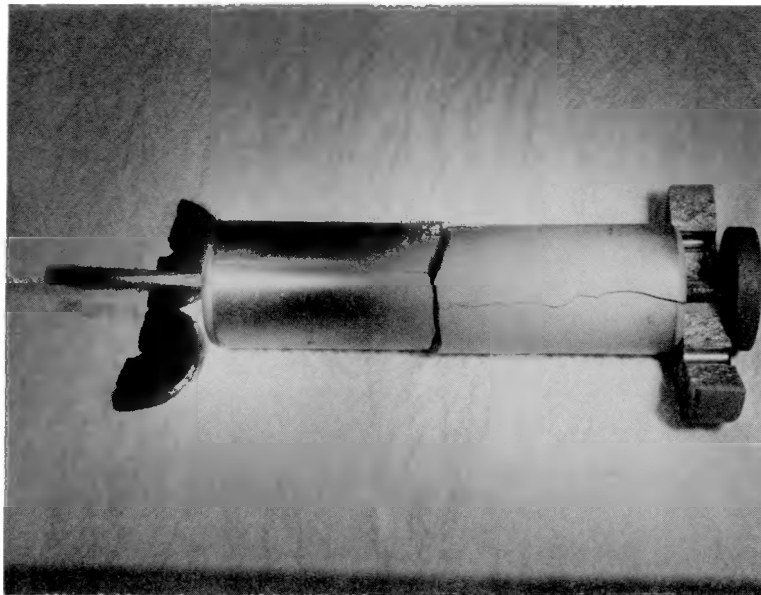


Figure 19. Photograph of $\text{U-ZrH}_{1.49}$ Specimens after 27 Pulses



Figure 20. Photograph of Fracture Surface of Lower Half of $\text{U-ZrH}_{1.49}$ Specimen

CHAPTER IX

TASK 9. DIAGNOSTIC SYSTEM

Activities on this task for this reporting period are described in Fast Reactor Safety Research Program Quarterly Report, October-December 1976, SAND77-0145, Reactor and Development Department, Sandia Laboratories, Albuquerque, New Mexico, March 1977.

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