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ANNULAR CORE PULSE REACTOR UPGRADE
QUARTERLY REPORT
January - March 1977

MASTER

Advanced Reactor Research Department

Prepared by Sandia Laboratories, Albuquerque,
New Mexico 87115 and Livermore, California 94500
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Nuclear Fuel Cycle Programs

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January - March 1977

Submitted by
Advanced Reactor Research Department
Sandia Laboratories, Albuquerque, New Mexico 87115

Approved:

Jack V Walker
Manager, Advanced Reactor Research

W. L. R. R. R.
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.⁴⁻⁹

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 $\text{U-ZrH}_{1.5}$ 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 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. Procurement of the clad, niobium liner, and insulators is part of this task.

Task 6. BeO-UO₂ 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-UO₂ fuel material. Since BeO-UO₂ was chosen as the fuel for the upgrade core, this task is concerned with production of the BeO-UO₂ 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-UO₂ 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-UO₂ 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 for review and evaluation in February, 1977. After the decision was made to use an all BeO-UO₂ core, major portions of the document were rewritten to reflect this decision. Chapter IV, Reactor, and Chapter XIV, Accident Analysis, were the chapters which required the greatest amount of revision.

Accident Analysis

Since a core consisting entirely of BeO-UO₂ fuel elements was chosen instead of a mixed core of BeO-UO₂/U-ZrH or (U,Zr)C-graphite/U-ZrH, the accident analysis was revised to reflect this decision. The reactor parameters applicable to the accident analysis are given in Table I.

A maximum nominal single pulse is estimated to require a reactivity insertion of \$2.70. The calculated pulse shape and energy release are shown in Figure 2. Without scram, the total energy release at 10 seconds is 390 MW second which corresponds to a peak fuel temperature rise of 1400°C. If the total worth (\$3.20) of the transient rod bank is inadvertently inserted, a pulse with a yield of 500 MW-second would result. A fuel maximum temperature rise of 1800°C would occur; the peak clad heat flux is calculated to be a factor of two below the film boiling heat flux so no clad damage would be expected.

TABLE I

Typical Reactor Parameters
Applicable to Accident Analysis

Steady-State Operation

Maximum power level	2 MW
Maximum BeO-UO ₂ fuel temperature	1200°C
Maximum heat flux	$2.8 \times 10^5 \text{ W/m}^2$
Minimum DNBR	6.1
Maximum convection velocity (core)	0.43 m/s
Axial peak/average power ratio	1.35
Radial peak/average power ratio	1.3
Power level trip value	2.2 MW
Power level trip delay*	2.0 s

Transient Operation

Maximum step reactivity insertion	\$2.70
Maximum power **	27,000 MW
Maximum energy release (pulse)**	390 MW/s
Maximum fuel temperature	1400°C
Axial peak/average power ratio	1.35
Radial/peak/average power ratio	1.3

Steady-State or Transient Operation

Moderator temperature coefficient of reactivity	- 0.007\$/°C
Fuel temperature coefficient of reactivity Average over a range of 27°C - 627°C	- 0.005\$/°C
Moderator void coefficient of reactivity	- 0.35\$/% void
Neutron lifetime	24 μs
Delayed neutron fraction	0.0073

Steady-State or Transient Operation

Control rod bank worth	\$12.00
Transient rod bank worth	\$ 3.20
Safety rod bank worth	\$ 2.50
Water height over core	7 m

* Assumed for analysis purposes, unless otherwise noted.

**\$2.70 step reactivity insertion.

Figure 2. Power and Energy Release for \$2.70 Pulse

Another accident which was examined was a continuous withdrawal of the control rod bank. The calculated behavior of the reactor for this accident is shown in Figure 3. The peak fuel temperature reached 2860°C and if this extremely unlikely event occurred considerable core damage could be expected.

In order to estimate the maximum consequences due to fission product release from the BeO-UO₂ core, a complete core disruptive accident was examined. The fission product inventory at the time of the accident was assumed to be that resulting from continuous operation of the reactor at 300 kW for 10 years, followed by 2 MW operation for 8 hours and the maximum pulse. It was assumed that 100% of the halogens and noble gases are released to the water, 99% of the halogens are retained in the water, and 1% of the halogens along with 100% of the noble gases are instantaneously released to the atmosphere through the building ventilation system.

Calculations using the RSAC¹¹ code were performed by Intermountain Technologies, Inc., of Idaho Falls, Idaho, with diffusion coefficients for Pasquill F conditions, a building wake factor of 0.5, a wind speed of 1 m/s, and no credit taken for filters. The doses which were calculated are given in Table II.

Initial Criticality Test (NST-1)

The test procedure for the Initial Criticality Test for the ACPR Upgrade (NST-1) has been written and reviewed and approved by the ACPR Committee. The purpose of the Initial Criticality Test is to safely load the reactor core with fuel material until a critical configuration is achieved.

Description of Test

All preoperational testing must be completed prior to the start of the Initial Criticality Test.

Fuel elements will be loaded into the core in a prescribed sequence with a fuel handling tool. The number of fuel elements to be loaded during any loading step will be based upon calculational predictions and continuously reevaluated reactivity monitoring. At the end of each loading step, the loading operation will be stopped and count rate data will be taken and evaluated before proceeding. Calculational predictions will be used to set the number of fuel elements that shall be loaded during the first few loading steps. Reactivity monitoring will be used to set the number of fuel elements to be loaded during each loading step for the remainder of the

Figure 3. Power and Temperature Histories for Continuous Control Rod Withdrawal

TABLE II

Unfiltered Doses for Complete Core Disruption

<u>Organ</u>	<u>TA-V Boundary 100 meters (Rem)</u>	<u>Exclusion Radius 3 kilometers (Rem)</u>	<u>General Population 6 kilometers (Rem)</u>
External Gamma	18.0	2.30	0.50
Lung	29.0	18.00	15.00
Thyroid	470.0	13.00	5.40
G. I. Tract	1.0	0.04	0.02
Bone	4.0	0.20	0.09
Whole Body	1.0	0.03	0.01

The doses listed under TA-V Boundary are considered the on-site doses and the 3,000 meter doses are those which must meet 10CFR50 requirements.

NST-1. Reactivity monitoring will be accomplished by the familiar inverse subcritical multiplication technique. The subcritical neutron multiplication factor (M) is defined as follows:

$$M = \frac{1}{1 - k_{\text{eff}}}$$

also

$$M \approx \frac{C}{C_0}$$

hence

$$\frac{1}{M} = 1 - k_{\text{eff}} \approx \frac{C_0}{C}$$

where

C = nuclear instrumentation count rate during a fuel loading step,

C₀ = initial nuclear instrumentation count rate (no fuel present).

Since the inverse multiplication factor is related to the amount of fuel loaded in the core, the critical loading can be predicted from measurements during prior loading steps. From the above relationship, C approaches infinity as k_{eff} approaches 1 (criticality). Hence, a plot of 1/M versus fuel loading can be used to extrapolate to the critical fuel loading; that is, the critical fuel loading is predicted to occur at the point where 1/M extrapolates to zero.

Core geometry may significantly affect the observed count rates; hence, the core will be loaded in a compact geometry (hexagon) and the source will be located in the center of the core. Further, to enhance the reliability of the 1/M plots, at least three independent low-level count rate channels will be installed to compliment the standard control console instrumentation. Scalars will be used with these additional channels.

Fuel accountability records will be maintained during this test. In addition, the procedures will require a verification check to assure that all fuel elements are loaded in their proper locations. The actual critical fuel loading must be within the design loading to meet the acceptance criterion. This criterion assures that all calculations relating to reactivity predictions are correct.

Additional 1/M data will be taken after attaining the critical loading for various control, transient, and safety rod insertions. These data will be used to provide preliminary rod worth estimates for NST-3.

It is intended that this test procedure will serve as a guide for the preparation of other test procedures.

CHAPTER II

Task 2. Core Nuclear Design

P. S. Pickard, 5420A

The preliminary core design calculations initiated last quarter for the single-region UO_2 -BeO core concept were completed. The detailed comparisons made between the single- and multiple-region core designs resulted in the selection of the all BeO- UO_2 core design for the ACPR Upgrade. A summary of these calculations is included in this report. The change in core design required a thorough reevaluation of calculations performed for the Safety Analysis Report (SAR). These SAR calculations were also completed during this quarter.

Single-Region Core Studies

The primary concern in utilizing a single-region, UO_2 -BeO-fueled core design was the reduced temperature feedback available from the BeO based fuel. The elimination of the two-region interface, however, considerably reduces fission density peaking which reduces fuel stress and allows higher performance. The survey calculations performed to evaluate the single-region core design for the ACPR Upgrade utilized the same design constraints and operational goals as the two-region core designs. Core size and fuel loading considerations are summarized in Table III. Core size was restricted by the current grid plate to ≤ 6 full rows of fuel. Fuel loadings were limited to less than 30 w/o UO_2 in BeO. The single-region core has higher relative worth in the outer region, and reflector elements are required to control fission density peaking and reduce the neutron generation time. Survey calculations examined graphite and stainless steel reflector elements.

Relative fission density profiles for 4-, 5-, and 6-row core designs are shown in Figure 4. The relatively small 4-row core (132 elements) shows considerable fission density peaking due to high fuel loading. The larger core designs show milder profiles indicating lower thermal stresses in the BeO fuel. Calculations were performed with two-dimensional S_N analysis with 9-group cross sections generated from ENDF/B. The actual fuel loading required for an excess reactivity of 6 percent is shown for all core designs in Figure 5. The low enrichment, small core designs require up to 16 w/o $^{235}\text{UO}_2$ when reflected with stainless steel elements. The larger high-enrichment core designs

TABLE III

Core Size and Fuel Loading for
Single Region BeO-UO₂ Core Designs

Core Size (# Elements)	w/o ²³⁵ UO ₂	w/o ²³⁸ UO ₂	w/o UO ₂	Reflector*
132	13.0	6.0	19	SS/G
132	14.8	14.0	28.8	G
180	6.5	6.0	12.5	SS/G
180	7.6	14.0	21.6	SS/G
234	4.5	6.0	10.5	SS/G
234	5.3	14.0	19.3	SS/G

*SS - Stainless Steel Dummy Elements

G - Graphite Dummy Elements

Figure 4. Relative Fission Density for Single-Region $\text{UO}_2\text{-BeO}$ Core Designs (Stainless Steel Reflector)

Figure 5. U^{235} Loading for Single-Region UO_2 -BeO Core

require only 6 w/o $^{235}\text{UO}_2$ using the stainless steel reflector elements. The range of fuel loadings considered for the two-region core designs was 9.5 to 11.5 w/o $^{235}\text{UO}_2$. Overall core performance (pulse fluence) increases with reduced fuel loading and local peaking effects are reduced. The combined effects of increased ^{238}U loading and stainless steel reflecting elements requires an increase of approximately 30 percent in ^{235}U concentration. Both of these design features are directed toward reducing pulse widths and initial periods.

The neutron energy spectrum for a one-region core is compared to a two-region core and the current ACPR in Figure 6. The reduced fuel concentration in the single-region core results in a slightly softer cavity spectrum than the two-region core. The neutron spectrum is still considerably harder than the current ACPR spectrum.

The pulse performance improvement factor for the variety of designs examined in the survey calculations is shown in Figure 7. The core designs with the softest reflector and highest enrichment have the lowest fuel concentrations and the highest performance potential for a given core size. It is apparent that core designs exceeding about 200 elements will be able to meet the performance goal of 2.6 at 1200°C maximum fuel temperature. These performance levels are approximately 10 percent higher than the two-region interface peaking effects.

Analysis of reactor kinetics was performed using two-dimensional S_N calculations to determine reactivity loss with temperature and Monte Carlo techniques to evaluate the neutron generation time. These results were used in a point kinetics model to estimate minimum initial periods and pulse widths. The results of the S_N calculations for the temperature coefficient of reactivity are shown in Figure 8. The temperature coefficients of reactivity shown are average values over the temperature range of 27°C to 627°C, which is the approximate average core temperature range for the BeO-UO_2 fuel. The magnitude of the temperature coefficient for the single-region cores is in the range of $-0.3 \rightarrow -0.6$ $\phi/^\circ\text{C}$ while the two-region core designs were in the range of $-0.6 \rightarrow -0.85$ $\phi/^\circ\text{C}$. The temperature coefficient for the current ACPR is -1.3 $\phi/^\circ\text{C}$. The longer initial periods and greater pulse widths indicated by the reduced temperature coefficient can be mitigated to some degree by a reduction in the neutron generation time. The hydride-based fuels provided a significant increase in the negative temperature coefficient, but the very low fissile concentration and soft neutron spectrum also resulted in a fairly long neutron generation time for the two-region cores. Values calculated for the two-region cores were in the range of 36-43 μs . Changes in reflector materials did not significantly reduce the generation time due to the relatively low worth at the outside edge of the two-region core. The neutron generation times for the

Figure 6. Relative Maximum Thermally Induced Stress in Fuel Pellets for One- and Two-Region Cones

Figure 7. Cavity Flux Spectrum Comparison

Figure 8. Performance Improvement Factor for the Single-Region $\text{UO}_2\text{-BeO}$ Core for a Maximum Fuel Temperature of 1200°C

single-region UO_2 -BeO cores were calculated using the Monte Carlo code KENO-II. The results are shown in Figure 9 for the variety of core designs examined in the survey calculations. The higher ^{238}U concentrations and harder reflector materials result in reduced generation times. The magnitude of this reduction as a function of design parameters is most significant for the smaller cores where the reflector influence is strongest. The combined effect of reflector, ^{238}U content core size, and fissile concentration on the minimum initial period are shown in Figure 10. The stainless steel reflected designs show the expected trend of initial period increasing with core size. For this approach, increasing core size causes a larger increase in neutron generation time than the accompanying increase in temperature feedback due to the reduction in fissile content.

The opposite trend is observed for the graphite-reflected core designs. The decrease in initial period with increasing core size is due to the influence of the soft reflector. The neutron generation time is dominated by soft reflector neutrons so that the increase in generation time is less significant than the increase in the temperature feedback due to more dilute fissile concentrations for the larger cores. The initial period, however, remains greater for the graphite reflectors. Figure 10 illustrates the diminishing effect of the reflector on reactor kinetics as the core size increases.

The combined effects of design parameters on kinetics and performance is shown in Figure 11. The solid lines represent an approximation for a continuous change in reflector properties from stainless steel to graphite. It is apparent that the performance goal of 2.6 requires a core size of about 5 rows for the slower core designs and about 5-1/2 rows for the faster core designs. For a maximum fuel temperature of 1200°C , the minimum initial period is only slightly longer than the two-region core design. All core designs had 6 percent excess reactivity and had a fuel height of 50.8 cm. A comparison point for the reference two-region core design is also shown on Figure 11. For 1200°C maximum fuel temperature in both cores, the single-region core is characterized by higher performance and slightly longer initial periods. The two-region reference core figure is for a maximum fuel temperature in the hydride region of 1000°C . Since it was not apparent that this temperature could be utilized in the hydride fuel, the performance and initial period for the two-region core designs might be somewhat less favorable. On the other hand, if hydride fuel was available that could be driven to temperatures in excess of 1000°C , the two-region core would appear more favorably in comparison with the single-region design. The same data are shown in Figure 12 for 1400°C maximum fuel temperature in the BeO- UO_2 fuel. At the higher temperatures the single-region core has considerably more performance with comparable

Figure 9. Temperature Coefficient of Reactivity for Single-Region UO_2BeO Core

Figure 10. Neutron Generation Times for Single-Region UO_2BeO Cores (Monte Carlo @ 27°C)

Figure 11. Initial Period for Single-Region UO_2 -BeO Cores for 1200°C Maximum Fuel Temperature

Figure 12. Performance Improvement and Initial Period for Single-Region UO_2 -BeO Cores

Figure 13. Performance Improvement and Initial Period for Single-Region Core ($1400^{\circ}C$)

initial periods. Since fuel stress during pulse operation is expected to be reduced by ≈ 30 percent, the potential of attaining higher temperature operation is enhanced for the single-region core.

Final design calculations were initiated for the single-region core which focused on a nominal 200-element core (5-1/3 rows) using nickel reflecting elements and 21.5 w/o UO_2 in BeO. Uranium enrichment has been fixed at 35 percent. These calculations formed the basis of the SAR reactor analysis sections and will be covered in the next quarterly report.

CHAPTER III

Task 3. ACPR Upgrade, Console Development

W. H. Sullivan, 5423

Designs of the linear power monitor, log power monitor, reactor shutdown channels, buffer amplifier and magnet power supply systems were completed. The final design review was completed for all these systems and the drawings were released for fabrication. Material lists were prepared and parts ordered; fabrication of these items should begin by March 31, 1977.

The electrical design for a variable speed stepping-motor driven rod drive system was completed and evaluated. Based upon the results of a preliminary design review and the laboratory evaluation, an extensive redesign of this system was undertaken and is now approximately two-thirds completed. The major concerns in this redesign effort are (1) to ensure stability of rod drive speeds, and (2) to provide system flexibility so that a number of proposed high-power operating modes can be easily installed at a later date. This design will be finalized by April 29, 1977.

Electrical design and laboratory evaluation of a peak-hold data retrieval system were completed. Preliminary design approval was granted for this system.

Preliminary design review of the percent power and peak power measurement systems was also completed. Design changes were recommended for both these systems and parts have been ordered in preparation for their laboratory evaluation.

CHAPTER IV

Task 4. Mechanical Design

H. C. Walling, 1136

Cooling System

The contractor has been selected for the construction of the ACPR Upgrade 2-MW cooling loop and the construction is scheduled to start October 3, 1977.

Penthouse Addition

Final design of the Penthouse addition to the ACPR high-bay area was completed. A request for quotation has been sent to a number of potential bidders.

Ventilation System

A detailed design of the cavity ventilation system has been completed and a request for quotation is now being prepared.

Bridge Support Structure

Detailed design for a new bridge structure for the ACPR Upgrade pool has been finalized and a purchase order for the structure is in preparation.

Coolant Inlet Manifold

A detailed design for modification of the ACPR tank liner has been initiated. The design should be completed in early April 1977.

Detector Housing

A detailed design for a replacement detector housing was begun during the past quarter. The replacement unit will allow precise control over detector instrument location adjacent to the ACPR upgraded core.

ACPR Tank Floor Modifications

The proposed Fuel Motion Detection Experiment shield will require some modification of the ACPR tank floor. A conceptual design of these modifications has been begun.

CHAPTER V

Task 5. Fuel Element Design

J. A. Reuscher, 5424; H. C. Walling, 1136

Introduction

This task involves the design and testing of the BeO-UO₂ fuel elements. The results of the in-pile fuel experiments were used to develop the fuel material and the fuel dual-annulus configuration. The major activities for this quarter involved the final fuel element design and heat transfer calculations for the Safety Analysis Report. The prototype niobium cups for the inner refractory liner of the fuel element were successfully fabricated.

Fuel Element Design

Beryllium Oxide - Uranium Oxide material, selected as the Upgrade fuel, consists of a BeO matrix with a small volume percentage of UO₂. The material is manufactured by cold-pressing and sintering techniques to 99+ percent of theoretical density. The UO₂ loading is approximately 21 weight percent which results in a volume fraction of UO₂ of 6.9 volume percent. The uranium is 35% enriched in U²³⁵. The physical properties of the BeO-UO₂ are listed in Table IV.

TABLE IV

Physical Properties of BeO-UO₂ Fuel

<u>Property</u>	<u>Value</u>
Thermal conductivity at 21°C	240 W/m °C
Elastic modulus:	
at 21°C	37.8 x 10 ¹⁰ Pa
at 1000°C	34.1 x 10 ¹⁰ Pa
at 1500°C	22.3 x 10 ¹⁰ Pa
at 2000°C	8.0 x 10 ¹⁰ Pa
Ultimate tensile strength	
at 21°C	2.07 x 10 ⁸ Pa
Density (g/cc)	3.55 (21 wt. % UO ₂)
Thermal expansion coefficient 100°C - 1000°C	8.5 x 10 ⁻⁶ /°C

The fuel element for the BeO-UO₂ fuel consists of an outer stainless steel cladding with a wall thickness of 0.5 mm and a total length of 58.93 cm (23.2 inch). The fuel is manufactured in dual concentric annuli with an inner annulus I.D. of 4.183 mm (0.19 inch), O.D. of 21.84 mm (0.866 inch), and height of 6.35 mm (0.25 inch). The outer fuel annulus has an I.D. of 22.35 mm (0.880 inch) and an O.D. of 33.65 mm (1.326 inch). The fuel element design is shown in Figure 13. The BeO-UO₂ dual annuli are slotted at 180 degrees along the length to reduce thermal stresses; then they are loaded, as shown on the figure, in niobium cans (which serve as a liner) with a height of approximately 10.16 cm (4 inch). The niobium cans are assembled to provide an active fuel region of 50.8 cm (20 inch), and are placed within the stainless steel cladding with a 1.90 cm (0.75 inch) thick BeO end piece at the top and bottom of the fuel stack. There is a gap between the niobium and clad of 0.51 mm (.020 inch), and a second gap of 0.51 mm (.020 inch) between the I.D. of the niobium and the O.D. of the fuel. Gap spacing at both locations is assured by longitudinal corrugations in both the inside and outside surfaces of the niobium cylinder. The niobium forms a refractory inner liner with a melting point of approximately 2400°C. The inner cans also provide additional safety margin in case a BeO-UO₂ fuel pellet suffers extensive cracking with prolonged use in the reactor.

The stainless steel end fittings are streamlined to reduce the resistance to the natural convection flow in the channel around the fuel element. The cladding structure surrounding the internal components of the fuel elements is made of type 304 stainless steel tubing with walls 0.51 mm thick. Type 304 stainless steel also is the base material for the end fittings to which the cladding is welded. In addition, the cladding is fabricated with a nominal 0.51-mm thick annular space between the cladding and the internal parts of the fuel element. The inert gas-filled space provides a thermal barrier between the high-temperature surfaces and the water coolant. The annular gap is maintained during expansion of the fuel by a shallow rib design included in the liner. The O.D. of the fuel element is nominally 37.47 mm (1.475 inch). At the upper end of the fuel element is a grooved extension designed to fit and lock into a fuel handling tool.

The heat capacity of the BeO 21 weight percent UO₂ fuel as a function of temperature is shown in Figure 14.

Figure 15. Heat Capacity of BeO-UO₂ Fuel

Heat Transfer Calculations

Steady-state and transient calculations were performed for the BeO-UO₂ fuel element design using TAC2D.¹² The steady-state parameters for 2-MW operations are given in Table V.

TABLE V

Typical Heat Transfer and Hydraulic Parameters For
200* Element ACPR Upgrade Core for 2.0-MW
Steady-State Operations

<u>Parameter</u>	<u>Value</u>
Diameter of Fuel Element	37.47 mm (1.475 inch)
Length (heated) of Fuel Element	50.8 cm (20 in.)
Cell Area	0.001506 m ² (2.3349 inch ²)
Flow Cross Sectional Area/Element	4.04 cm ² (0.626 inch ²)
Wetted Perimeter/Element	117.7 mm (4.634 inch)
Hydraulic Diameter/Element	13.73 mm (0.5405 inch)
Heat Transfer Surface/Element	0.060 m ² (92.7 inch ²)
Coolant Temp. Rise Through Core (Max) °C	12°C
Coolant Mass Flow/Channel (Avg.)	650 kg/hr
Peak Fuel Temperature (BeO-UO ₂)	1200°C Typical
Average Heat Flux	1.6 x 10 ⁵ W/m ²
Maximum Heat Flux	2.8 x 10 ⁵ W/m ²
Minimum CHF Ratio (MCHFR)	6.1

*Includes 6 fuel-followed control rods and 2 fuel-followed safety rods.

The maximum and average fuel temperature as a function of fuel element power is shown in Figure 15. The closure of the gaps between fuel and niobium liner and between liner and clad with increasing temperature is included in the calculations. This closure of the insulating gaps results in increased heat transfer at higher powers and produces the nonlinear rise in fuel temperature with power.

For transient mode calculations, the maximum clad heat flux is shown as a function of peak fuel energy deposition in Figure 16. These calculations included the closure of the insulating gaps. Since the departure from nucleate boiling heat flux is 5.4×10^5 Btu/hr - $^{\circ}\text{R}$, these results indicate that the pulse mode of operation is at least a factor of two below DNB. For the \$3.20 accident pulse discussed in Chapter I, the peak energy deposition is about 750 cal/g ($\sim 1800^{\circ}\text{C}$) and the heat flux is 2.7×10^5 Btu/hr - $^{\circ}\text{R}$ which is a factor of two below DNB.

The calculated peak clad temperature is given in Figure 17 for both pulse and steady-state operation. The gap closure model was utilized in these calculations. For both modes of operation, the clad temperature is expected to remain below 150°C , except for a high steady-state power of 18 kW per element. These results indicate that only modest clad temperatures are expected for the BeO-UO_2 fuel element.

Figure 16. Fuel Temperature Versus Element Power

Figure 17. Peak Clad Heat Flux Versus Energy Deposition

Figure 18. Peak Clad Temperature Versus Steady-State Power and Versus Pulse Energy Deposition

CHAPTER VI

Task 6. Fuel Element Fabrication

C. H. Karnes, 5835

It was determined early in this reporting period that an all UO_2 -BeO core was not only feasible but also resulted in improved performance over the two region core using the hydride driver fuel. The fabrication capabilities of LLL and IASL were determined to be sufficient to produce the 225 fuel elements in the required time period and most of the fabrication details and schedules have now been defined.

Fuel Fabrication

An order was placed at LLL for the entire quantity of fuel totaling 72,900 individual BeO- UO_2 pieces. The pieces are to be cold-pressed and sintered to final dimensions with no machining. An automatic press is being set up to produce the large number of pieces. The fuel pieces will be pressed at 30,000 psi from a mixture of BeO, U_3O_8 and 3 percent polyethylene-glycol binder. The fuel pieces will be sintered and reduced in hydrogen for two hours at 1650°C . A quality assurance program has been defined which includes chemical analysis at various stages, enrichment, 100 percent dimensional inspection, immersion density sampling, visual inspection for cracks, and a random test for flatness.

Three contract personnel from EG&G will begin working at LLL on April 4 to facilitate the high fabrication rate required. The fabrication system will be committed to $^{238}\text{UO}_2$ -BeO during the week of April 11 at which time material will be produced for a prototype fuel element.

The fabrication of enriched fuel is scheduled to begin on May 9 and, since each fuel element requires three different sizes of fuel pieces (outer, minimum, outer maximum and inner), the following schedule has been defined.

Start	Finish	Configuration	Fabrication Time-Weeks	No. Of Pieces	Supports No. Of Completed Elements
5/9/77	5/23/77	Outer Min.	2.0	4,500	225
5/27/77	7/18/77	Outer Max.	7.0	15,680	112
7/22/77	9/28/77	Inner	9.6	21,440	134
10/4/77	11/21/77	Outer Max.	7.0	15,820	113
11/28/77	1/19/78	Inner	6.6	14,560	91

This schedule will result in LASL having sufficient numbers of all three sizes of fuel pieces to begin assembly by August 4. Details of the schedule reveal that all fuel for the 196 elements required for the full core will be completed by January 6, 1978.

Other Components

The deep drawn niobium cups into which the fuel pieces are assembled are to be fabricated by Osborn Machine Co., Albuquerque. The drawing process has been developed and a few high quality 2 $\frac{1}{4}$ -inch deep cups have been drawn to demonstrate the process. Since the fuel element requires deeper cups, five 4 $\frac{1}{4}$ -inch prototype cups are to be produced by May 15 for the prototype element. The cups are to be made from 99.83 percent niobium sheet supplied by Kawcki Berylco. One hundred twenty-five are to be available by July 1 and then 200 per month will be delivered until 1125 are finished.

The cast 304 SS fuel element end fittings and the 304 SS cladding tubes are being supplied by General Atomics. The end fitting to be used is the new streamlined design to enhance cooling water flow.

The BeO end pieces which serve as thermal insulators and neutron reflectors have been designed. They are to be made by Ceradyne Corporation by cold pressing and sintering to a density of 2.85 g/cc.

Detailed information concerning the dimensions of these components may be obtained from Sandia Drawing P59525.

Fuel Element Assembly

The component fabrication schedule calls for all of the cladding tubes, end caps, BeO reflectors, and sufficient quantities of the three sizes of fuel pieces to arrive at LASL so that assembly can begin on August 4. The fabrication schedule calls for LASL to assemble, weld, and certify approximately eight elements per week until December 7, 1977, at which time the rate must increase to 11 per week. On January 26, 1978, 196 fuel elements will have been completed. This assembly schedule is coordinated closely with LLL's fuel fabrication schedule.

LASL and Sandia are in the process of defining the weld requirements. These along with the assembly quality assurance program will be finalized during the next quarter.

CHAPTER VII

Task 7. Secondary Fuel Materials Studies

Since (UC-ZrC)-graphite was not chosen as a fuel, this task is no longer a part of the project (see Introduction to this report).

CHAPTER VIII

Task 8. Driver Core Fuel Element

J. A. Reuscher, 5450; F. M. Morris, 5452

The ACPR Upgrade core was originally conceived as a two-region core with uranium - zirconium hydride fuel elements as the outer region. Since the decision was made to utilize an entire core of BeO-UO_2 fuel elements, this task was dropped from the project. Further in-pile tests are scheduled with U-ZrH samples which have hydrogen-to-zirconium ratios varying from less than 1.5 to greater than 1.6. The results of these experiments will be reported in a future report separate from the ACPR Upgrade Quarterly Reports.

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, January - March 1977, SAND77-0611, Reactor Research and Development Department, Sandia Laboratories, Albuquerque, New Mexico, May 1977.

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