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# CAPABILITIES OF THE ANNULAR CORE PULSE REACTOR FOR REACTOR SAFETY EXPERIMENTS*

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## ABSTRACT

The pulse and steady state performance of the Annular Core Pulse Reactor (ACPR) is being improved to provide a facility for fast reactor safety experiments. A new two region core is being designed with a high-heat capacity fuel region around the central irradiation cavity and an outer region of uranium-zirconium hydride fuel. The new core is expected to operate with an initial period of 2 to 3 milliseconds and a cavity fluence that is 2 to 2.5 times greater than the current value.

## INTRODUCTION

The Annular Core Pulse Reactor (ACPR) is a TRIGA type reactor which has been in operation at Sandia Laboratories since 1967. The reactor is utilized in a wide variety of experimental programs which include radiation effects, neutron radiography, the effects of rapid energy deposition on fissile materials and fast reactor safety. During the past several years, the ACPR has become an important experimental facility for the United States Fast Reactor Safety Research Program and questions of interest to the safety of the LMFBR are being addressed. In order to enhance the capabilities of the ACPR for reactor safety experiments, a project to improve the performance of the reactor was initiated. It is anticipated that the pulse fluence can be increased by a factor of 2.0 to 2.5 utilizing a two-region core concept with high heat capacity fuel elements around the central irradiation cavity. In addition, the steady-state power of the reactor will be increased by about a factor of three.

## DESCRIPTION OF PRESENT ACPR

The main feature of the ACPR is the 22.86 cm diameter dry experimental cavity at the center of the core. The total neutron fluence available for the maximum pulse is $2.2 \times 10^{15}$ neutrons/cm$^2$. The present core is a heterogeneous assembly of stainless steel clad fuel elements which contain a zirconium hydride

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moderator homogeneously combined with 20% enriched uranium fuel. The fuel contains 12 weight percent uranium with a hydrogen-to-zirconium ratio of 1.62. For a maximum pulse a reactivity insertion of $4.50 is required and the peak adiabatic fuel temperature is about 900°C.

## UTILIZATION OF PRESENT ACPR

The use of the present ACPR for fast reactor safety research has been discussed by Coats, et.al.[1] Prompt burst excursion (PBE) experiments have been performed by Schmidt[2] on single, stainless steel clad, $UO_2$ fuel pins in sodium. Energy depositions in 20 percent enriched $UO_2$ have reached 3200 J/gm with a suitable moderator around the experiment. The fission energy deposition profile in these experiments showed a greater surface-to-centerline depression ratio than would be the case for an LMFBR. Post accident heat removal (PAHR) studies are being planned by Rivard, et.al.[3] These PAHR experiments can be performed using the intrinisc fission heating of a $UO_2$/sodium debris bed to about one percent of typical average operating fuel power. Fuel coolant interaction (FCI) and effective equation of state (EOS) experiments can also be performed in the present ACPR configurations.

## LIMITATIONS OF PRESENT ACPR

The primary limitation on the utility of the ACPR in LMFBR Safety studies is the degree to which the present facility approximates LMFBR conditions. Due to a relatively soft spectrum, PBE experiments are effectively limited to single pin geometries. With multi-pin geometries, self-shielding effects occur and preclude adequate energy deposition in the central pin. An increase in the total pulse fluence with a significant hardening of the spectrum is required in order to improve deposition profiles or to increase total energy deposition for multi-pin PBE experiments. An increase in steady state flux by a factor of three allows the simulation of post-accident heat removal conditions of about 3 percent of typical average operating fuel power.

## ACPR UPGRADE

The ACPR Upgrade is a modification to an existing facility and it is anticipated that most of the present reactor structure can be utilized. The present core occupies about one-half of the fuel element positions in the grid plate and the new core is designed to be compatible with the existing grid configuration; however, other modifications are anticipated for the reactor systems. A completely new control system will be installed, a 2 megawatt steady-state pool cooling capability will be installed, and the need for an improved containment or confinement system is anticipated because of the increased risk potential. Another part of the fast reactor safety research program at Sandia Laboratories is the investigation of techniques to observe the motion of molten fuel and clad during and after a reactor pulse. Such a device requires minor changes to the grid configuration so that a radial row of fuel elements can be removed to allow observation of an experiment in the cavity. Core design considerations for a fuel motion detection system are a part of the upgrade project.

The present ACPR core grid and experiment access facilities will not be changed for the improved core. The present $U-ZrH_{1.62}$ fuel elements are 3.56 cm in diameter and contain a fuel length of 38.1 cm with an 8.89 cm long graphite cylinder at each end of the fuel. The high heat capacity fuel elements will have the same clad outer dimensions as the present fuel elements; however, the graphite cylinders will not be employed and the fuel length will be increased to 50.8 cm. Three rings of high heat capacity fuel elements will be placed around the central cavity. The outer core region will consist of three rings of $U-ZrH_{1.5}$ fuel elements which will provide an adequate negative temperature coefficient for the core.

## HIGH HEAT CAPACITY FUELS

A number of high heat capacity fuels which have potential for pulsed reactor applications have been discussed by Sasmor, et.al.[4] Two fuels, $BeO-UO_2$ and (UC-ZrC)-graphite; were chosen as the high-heat capacity fuel candidates for the ACPR Upgrade. Prior to this project, there was little information concerning the behavior of these fuels in a pulsed reactor environment. An extensive fuel investigation program has been conducted using the present ACPR to develop and test the high heat capacity fuel element design. The choice of which fuel to use with the improved core depends heavily on the results of the fuels testing program. Samples of $BeO-UO_2$ are provided by Lawrence Livermore Laboratory and the UC-ZrC)-graphite materials are supplied by Los Alamos Scientific Laboratory.

A major part of the fuel testing program is directed at minimizing the damage to the candidate fuel pellets with a diameter of 3.3 cm. The specimens are pulse heated in the ACPR to the temperature range required for use in the upgraded reactor and the post test condition is examined to determine the degree of cracking. The cracking is caused by large thermal stresses induced by a peak-to-minimum temperature ratio as large as 2.0. Various fuel fabrication techniques and geometry changes have been examined as stress reduction factors. The fuel material development programs are discussed in detail in the project quarterly reports.[5,6] In addition, the $BeO-UO_2$ fuel development has been summarized by Pickard, et.al[7]; and the (UC-ZrC)-graphite development program has been described by Marion, et.al.[8]

The fabrication processes being employed are cold pressed and sintered with the $BeO-UO_2$ and hot pressed and extrusion for the (UC-ZrC)-graphite. The fuel tests have shown that a dual-slotted annulus configuration can be used with both $BeO-UO_2$ and extruded (UC-ZrC)-graphite to reduce thermal stresses and improve the survival of an individual specimen. For the hot pressed (UC-ZrC)-graphite, a solid cylinder is able to survive the in-pile tests. Repetitive tests in excess of 100 pulses are performed to examine the effect of temperature cycling on the fuel samples.

The fuel testing results are being used to develop a fuel element design that is compatible with the high operating temperature requirement up to 1400°C for the $BeO-UO_2$ and 2000°C or above for the (UC-ZrC)-graphite. The fuel element design requires an insulating liner between the fuel and clad to prevent excessive clad temperatures. The liner materials are niobium for the

$BeO-UO_2$ fuel element and graphite for the (UC-ZrC) graphite fuel element. In-pile testing of prototype fuel element designs are being conducted. The details of the high heat capacity fuel element designs and the in-pile test results are discussed in the project quarterly reports.[5,6,14] In addition, other in-pile experiments are examining the behavior of the outer core region fuel, $U-ZrH_{1.5}$, up to temperatures of 1300°C.

## CORE PHYSICS ANALYSIS

Since the operational fuel temperatures for the high heat capacity fuels were not firmly established, survey calculations for the core physics were performed for a range of maximum fuel temperatures. The upgraded ACPR will be required to accommodate a range of large negative worth experiments, so the amount of excess reactivity was also parameterized to examine a range of fuel operating temperatures and excess reactivity requirements. Cross section generation for the large variety of water/fuel cells required for these studies utilized ENDFB data and the AMPX cross section code system developed at Oak Ridge National Laboratory. The survey calculations focused on performance improvement as a function of the uranium concentration in the high heat capacity region (5 to 15 wt % $UO_2$ for the $BeO-UO_2$ and 200 to 800 mg U/cc for the (UC-ZrC)-graphite) with maximum fuel temperature, excess reactivity, and hydride region enrichment as parameters. A comparison between the present and future ACPR operational characteristics is given in Table I. The upgrade calculations were performed with the two different high heat capacity fuel regions and the $U-ZrH_{1.5}$ outer region. The maximum pulse temperatures of 2000°C for the (UC-ZrC)-graphite fuel and 1200°C for the $BeO-UO_2$ fuel represent comfortable design limits based on the results of the in-pile fuels testing program. The major difference between the operational characteristics of the two fuels is the pulse temperature. Factors other than performance greatly influence the choice of the high heat capacity fuel; these factors are cost, schedule and safety considerations.

Table 1

Present and Future ACPR Operational Characteristics

| Parameter | Present ACPR | UC-ZrC-C | $BeO-UO_2$ |
|---|---|---|---|
| Max Steady State Power | 600 kw | 2 MW | 2 MW |
| Max Steady State Flux ($n/cm^2 sec$) | $1.3 \times 10^{13}$ | $3.1 \times 10^{13}$ | $3.2 \times 10^{13}$ |
| Pulse $\Delta T$ (°C) | 900°C | 2000°C | 1200°C |
| Pulse Fluence ($n/cm^2$) | $2.2 \times 10^{15}$ | $4.6 \times 10^{15}$ | $5.0 \times 10^{15}$ |
| Min. Initial Period (msec) | 1.3 ms | ≃ 2.0 ms | ≃ 1.9 ms |
| Pulse width (FWHM, msec) | 4.5 ms | 7 ms | 6.7 ms |

The effect of both high heat capacity fuels on several typical fast reactor safety experiments has been analyzed. Due to the space limitations of this paper, only some representative calculations are discussed.

## SINGLE PIN PROMPT BURST EXCURSION EXPERIMENT

A single pin prompt burst excursion experiment was chosen for comparison calculations between the present and upgrade cores. A 20% enriched $UO_2$ fuel pin with a diameter of 0.495 cm was surrounded with a 0.32 cm thick stainless steel container. Various thicknesses of polyethylene were placed around the container and two-dimensional transport calculations performed with the assembly in the ACPR central cavity. This configuration is similar to the experiments conducted by Schmidt [2] in the present ACPR and was not optimized for the new core. The calculated results are given in Figure 1 for average energy deposition as a function of moderator thickness for a maximum yield pulse. The upgrade core for these calculations contained a (UC-ZrC)-graphite central region. Even though there is about 2.0 times the cavity fluence available with the new core, there is no increase in the unmoderated energy deposition. This is due to the harder neutron energy spectrum in the upgrade core; this results in a flatter energy deposition profile across the pin. It is necessary to moderate a single $UO_2$ pin in both the present and new ACPR cores in order to achieve fuel vaporization and, hence, pin failure. The reactor can be operated in a double pulse mode to establish an initial temperature profile across the pin. The second pulse should provide sufficient energy deposition to vaporize the $UO_2$ and fail the fuel pin. The present reactor is unable to create pin failure in the double pulse mode; however, the upgrade core does provide this capability.

## SEVEN PIN PROMPT BURST EXCURSION EXPERIMENT

A seven pin bundle of 0.495 cm diameter $UO_2$ fuel pins was modeled with a stainless steel container (0.32 cm thick) and surrounding moderator. A comparison calculation between the present and upgraded cores is shown in Figure 2. The energy deposition profile for a maximum yield pulse is shown for the current reactor with 1.27 cm of polyethylene around the experiment and for the upgrade core with 0.64 cm and 1.27 cm of polyethylene.

The central pin contained 40% enriched $UO_2$ in order to provide sufficient energy deposition to fail the pin. The experiment design was not optimized for the upgrade and was chosen only to show a relative comparison between the current and future ACPR cores.

The curves in Figure 3 illustrate the effect of moderator thickness on the average energy deposition in the central pin for a maximum yield pulse. For the present ACPR it is not possible to fail the central 40% enriched pin. The calculations indicate that central pin failure (depositions greater than 500 cal/gm) can be expected with upgraded core.

## DEBRIS BED EXPERIMENTS

Debris bed experiments at Sandia Laboratories are designed to provide information on the behavior of reactor materials following a hypothetical core-disruptive accident. The details of these experiments can be found in the research program quarterly reports 3,9-13 . One of the experiments was chosen for comparison calculations: a debris bed of small, $UO_2$ particles (10 to 1000 microns) in liquid sodium. The bed consisted of 52% fully enriched $UO_2$ and 48% sodium in a stainless steel vessel. The bed diameter was 10.16 cm and the height was 17.15 cm; numerous insulating, filtering and helium cooling regions were exterior to the vessel. These experiments are conducted at steady state and are designed to characterize sodium dryout instabilities in a bed with intrinsic heat generation. The heat generation rate in the $UO_2$ is shown in Figure 4 as a function of bed radius for the present ACPR at 600 kilowatts and the upgraded reactor at 1.5 and 2.0 megawatts. The new ACPR core is expected to provide up to a factor of three improvement in heat generation rate with a flatter deposition profile.

## CONCLUSION

The experiment comparison calculations show that a two-region core for the ACPR will greatly enhance the applications of this reactor for fast reactor safety experiments. Significant results have been obtained with the development of high-heat capacity fuels for pulse reactor applications. The fuels program for the ACPR Upgrade has shown that BeO $UO_2$ can survive pulse environments to 1200°C and (UC-ZrC)-graphite can be pulsed in excess of 2000°C. The critical experiment for the new core is scheduled for March, 1978, and the reactor is expected to be operational for experiments by June, 1978.

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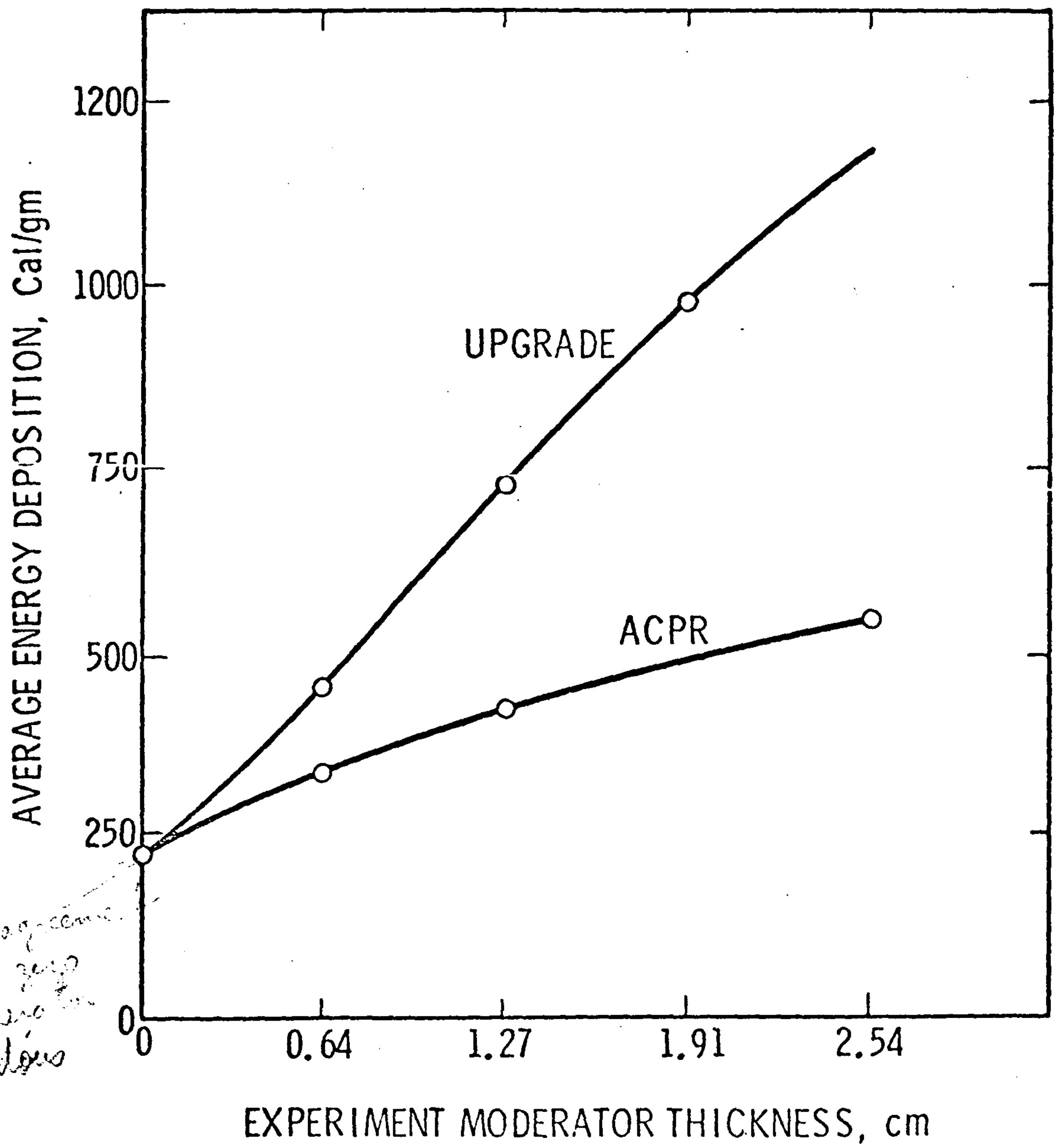
ENERGY DEPOSITION IN SINGLE $UO_2$ PIN (20% enriched)

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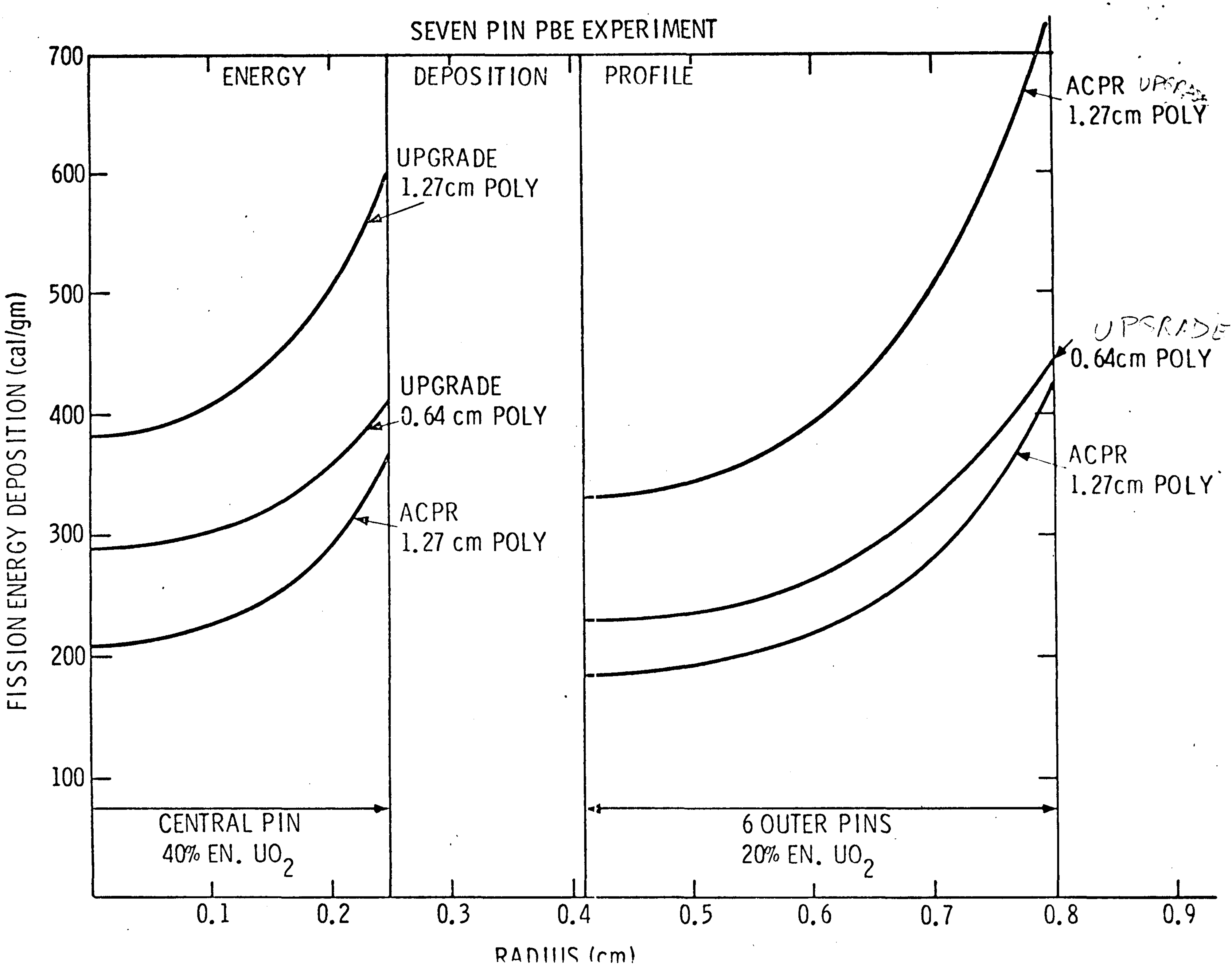
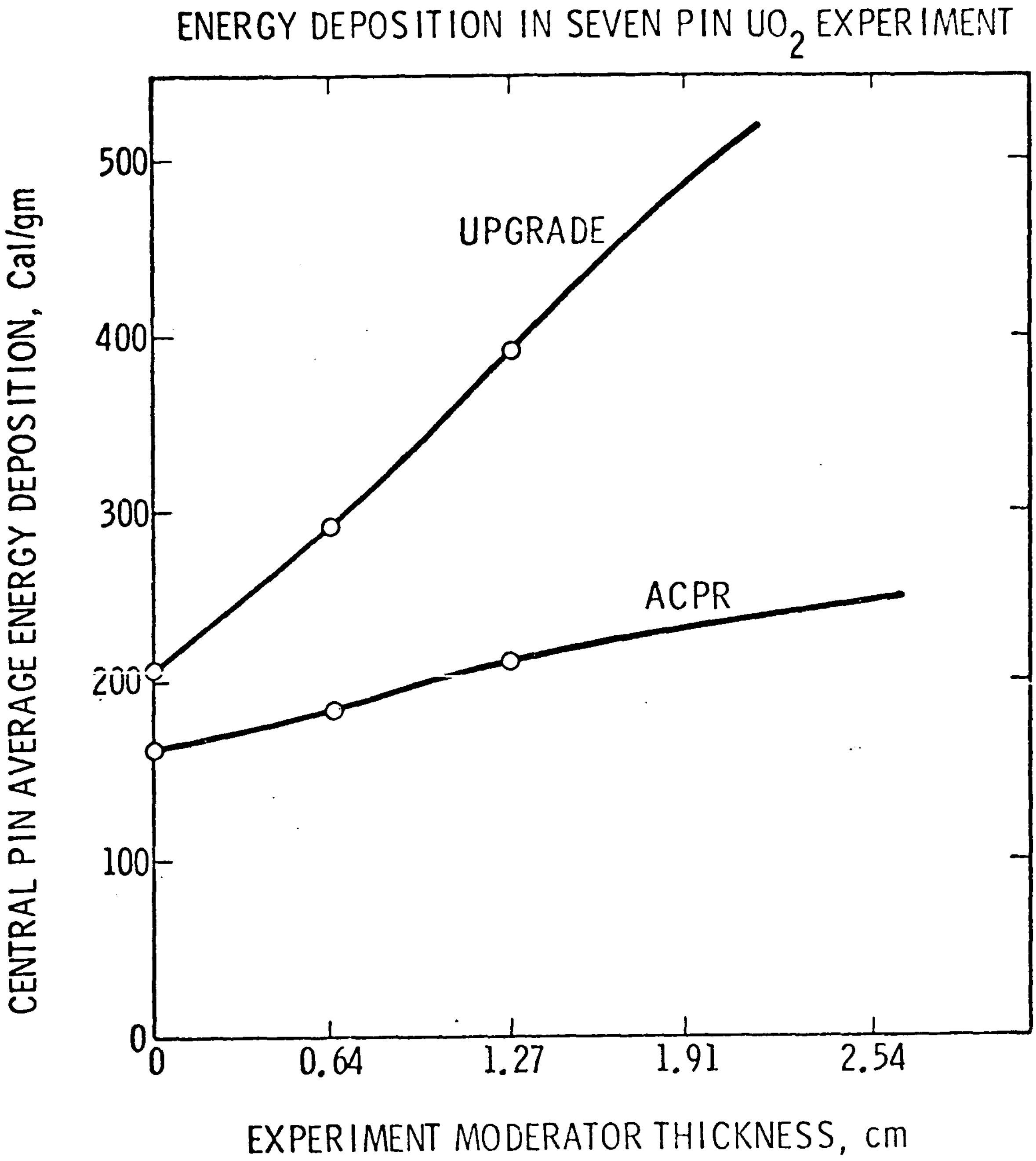

SEVEN PIN PBE EXPERIMENT
ENERGY
DEPOSITION
PROFILE
FISSION ENERGY DEPOSITION (cal/gm)
RADIUS (cm)
UPGRADE
1.27cm POLY
UPGRADE
0.64 cm POLY
ACPR
1.27 cm POLY
ACPR
1.27cm POLY
0.64cm POLY
ACPR
1.27cm POLY
CENTRAL PIN
40% EN. UO2
6 OUTER PINS
20% EN. UO2
UPGRADE
UPGRADE
700
600
500
400
300
200
100
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9


ENERGY DEPOSITION IN SEVEN PIN UO2 EXPERIMENT
UPGRADE
ACPR
500
400
300
200
100
0
0
0.64
1.27
1.91
2.54
CENTRAL PIN AVERAGE ENERGY DEPOSITION, Cal/gm
EXPERIMENT MODERATOR THICKNESS, cm

FISSION DENSITY PROFILE IN 52% $UO_2$ - 48% Na DEBRIS BED