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Performance Improvement of the Annular Core Pulse Reactor for Reactor Safety Experiments*

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Summary

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, activation analysis, 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 two.

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 3×10^{15} neutrons/cm^2. The maximum pulse results in an energy release of 108 Mw-sec with an initial period of 1.3 msec, and a pulse width at one half maximum power of 4.5 msec. The reactor can be operated at a steady state power of 600 kw, which produces a cavity neutron flux of 2×10^{13} neutrons/cm^2 sec. The present core is a heterogeneous assembly of

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stainless steel clad fuel elements which contain a zirconium hydride moderator homogeneously combined with 20% enriched uranium fuel. The fuel contains 12 weight percent uranium and 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.

Studies have identified several potential approaches to the ACPR performance improvement. The most promising approach appears to be a two-region core concept. The inner region, surrounding the irradiation cavity, would consist of a high-heat capacity fuel capable of absorbing the fission energy associated with a large nuclear pulse. The outer region would consist of a U-ZrH fuel to provide an adequate negative temperature coefficient for the two-region core. The high heat capacity fuels under investigation are BeO-UO_2 and (UC-ZrC)-graphite.

The present ACPR is utilized for a variety of fast reactor safety experiments. Prompt burst excursion (PBE) experiments have been performed by Schmidt(1) on single, stainless steel clad, UO_2 fuel pins in sodium. Energy depositions in 14 percent enriched UO_2 have reached 2700 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.(2) These PAHR experiments can be performed using the intrinsic fission heating of a UO_2/sodium debris bed up 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.

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, strong self-shielding effects occur and produce severe nonuniform heating profiles. An increase in the total pulse fluence combined with either a significant hardening of the spectrum or low energy neutron filtering is required in order to simultaneously provide energy deposition and energy deposition profiles adequate for the desired range of multi-pin PBE experiments. An increase in steady state flux of a factor of two allows the simulation of post-accident heat removal conditions of about

2 percent typical fuel decay powers.

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 a 8.89 cm long graphite cylinder at each end of the fuel. The high heat capacity fuel elements will have the same outer dimensions as the present fuel elements; however, the graphite cylinders will not be employed and the fuel length is expected to be 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. It is anticipated that U-$ZrH_{1.5}$ can be operated up to 1300°C.

Since the operational fuel temperatures for the high heat capacity fuels are 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 500 mg U/cc for the (UC-ZrC)-graphite) with maximum fuel temperature, excess reactivity, and hydride region enrichment as parameters.

Prior to the ACPR upgrade project there was little or no information concerning the behavior of BeO-UO_2 and (UC-ZrC)-graphite fuels in a pulsed reactor environment. An extensive fuel investigation program is being 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 will depend heavily on the results of the fuels testing program. Samples of BeO-UO_2 are being provided by Lawrence Livermore Laboratory and the (UC-ZrC)-graphite materials are being supplied by Los Alamos Scientific Laboratory.

A major part of the fuel testing program is directed at minimizing the damage to the candidate fuel materials under rapid fission heating. A more reliable fuel element design will result if the fuel material remains in tact. The in-pile testing program utilizes 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 the large induced thermal stresses and 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.[3,4] In addition, the (UC-ZrC)-graphite fuel development is described by Marion, et al.[5] 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. The three fabrication processes being employed are cold pressing and sintering, hot pressing, and extrusion. Multiple pulse tests are being performed to examine the effect of temperature cycling.

The fuels testing results are being used to develop a fuel element design that is compatible with the high operating temperature requirement up to 1500°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 molybdenum or niobium for the $BeO-UO_2$ fuel element and graphite for the (UC-ZrC)-graphite fuel element. In-pile testing of prototype fuel elements will be conducted. In addition, in-pile experiments will examine the behavior of the outer core region fuel, $U-ZrH_{1.5}$, up to temperatures of 1300°C.

The ACPR upgrade is a modification to an existing facility and it is anticipated that most of the existing reactor structure can be used for the upgrade. The present core occupies about one-half of the fuel element positions in the grid plate. Other modifications are anticipated for the reactor systems. A completely new control system will be installed, a larger pool cooling capability is required for a higher steady-state power, and the need for an improved containment or confinement system is being investigated and 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 may require 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.

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