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DEVELOPMENT OF A (U,Zr)C-GRAPHITE PULSED REACTOR FUEL ELEMENT*

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A new core is being designed to improve the performance of the Annular Core Pulse Reactor (ACPR). This design utilizes a two region core with the inner region containing high enthalpy fuel elements around the 228 mm diameter central irradiation cavity. An outer core region of uranium-zirconium hydride fuel elements is placed around the high-enthalpy inner core region to provide an adequate negative temperature coefficient for the entire core. A program has been conducted to develop a high-enthalpy fuel element using a graphite matrix fuel containing (U-Zr)C solid solution particles.

Since the (U,Zr)C-graphite fuel is subjected to a reactor period of two milliseconds with a large temperature gradient across the fuel diameter, fracture due to thermal stresses is the primary failure mode. The development program optimized this class of fuel for pulsed reactor application. A variety of graphite-based fuels were fabricated by the Materials Technology group of the Los Alamos Scientific Laboratory, and the details of this fabrication has been previously reported. (1,2) Fuel specimens were fabricated by both extrusion and hot pressing using low thermal expansion graphite flour with uranium densities from 345 to 800 mg/cc. Mechanical and thermophysical characterization studies were performed to determine elastic modulus, fracture strength, thermal expansion, enthalpy and thermal conductivity in order to perform thermal and structural analyses. Fuel specimens with a diameter of 33 mm were tested without fracture in the ACPR to peak temperatures in excess of 2000°C with radial peak-to-minimum energy deposition profiles of 2.0 to 3.0.

The outer diameter of the fuel element was fixed at 37.3 mm to be compatible with the existing core grid configuration. Since the (U, Zr)C-graphite must operate at high temperatures (2000°C to 2300° C) in the pulse mode an insulating liner is required to prevent excess temperatures in the 0.51 mm thick stainless steel clad. A filament wound graphite sleeve was chosen as the insulating liner because of its mechanical and thermal characteristics. The sleeve is one millimeter thick with an inside diameter of 34.0 mm. The thermal behavior of the fuel element design was analyzed with two dimensional heat transfer calculations. The fuel element design was verified by assembling a section of cladding with a 127 mm long fuel region and graphite liner. A water region was placed around the clad container

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and the apparatus was pulsed in the ACPR to peak fuel temperatures from 6000° C to 2300° C. The fuel and clad temperatures were measured; the clad temperature rise varied from 500° C to 120° C.

The program has successfully developed a (U, Zr)C-graphite fuel element with a high volumetric enthalpy which can be used in a water cooled pulse reactor. Extensive tests have demonstrated the feasibility and safety of the fuel element design.

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References

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