



Testing fast reactor fuels in a thermal reactor

Pavel Medvedev*, Steven Hayes, Samuel Bays, Stephen Novascone, Luca Capriotti

Idaho National Laboratory, P. O. Box 1625, Idaho Falls, ID 83415, USA

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ABSTRACT

The present lack of a domestic fast neutron flux irradiation capability combined with continued development of fast reactor fuels in the U.S. motivated an innovative engineering solution to utilize a unique neutron flux tailoring capability in the Advanced Test Reactor at the Idaho National Laboratory. To achieve the objectives of the fast reactor fuel irradiation tests, the incident neutron flux was hardened substantially by placing fueled irradiation capsules inside specially designed cadmium shrouds. Use of cadmium prevents thermal neutrons from reaching the fuels being tested and alleviates the plutonium self-shielding that would normally arise during irradiations of high density, highly enriched fuels in a thermal neutron spectrum.

The present paper illustrates the profound effect this engineered solution has on the efficacy of the experiments. Based on the comparison of post-irradiation measurements of the columnar grain region in fast reactor mixed-oxide fuels with fuel performance calculations, it is demonstrated that thermal conditions achieved in these cadmium-shrouded fuel experiments are substantially prototypic of a sodium fast reactor and are suitable for concept-screening tests supporting development of new fast reactor fuels. It is also shown that if the experiments were conducted in an unmodified ATR neutron spectrum, gross plutonium self-shielding would cause a strong depression of the fission power at the fuel centerline preventing fuel restructuring, a hallmark feature of mixed oxide fuel behavior under fast reactor conditions.

Recognizing the need for testing metallic fuels for fast reactors, the impact of neutron flux energy spectrum on the radial temperature distribution in metallic fuel is investigated. It is shown that the use of Cd shrouds allows to attain radial temperature distributions nearly identical to those that exist in an SFR.

1. Introduction

Since 2003 Idaho National Laboratory (INL) has been conducting irradiation testing of metallic and oxide fuels intended for in-reactor transmutation of nuclear waste (Hayes et al., 2016). This work is performed with an objective to develop sustainable nuclear fuel cycles as outlined in the DOE Nuclear Energy Research and Development Roadmap (U.S. Department of Energy, 2010). It is expected that current research and development efforts will culminate in a suite of options that will enable future decision makers to make informed choices about how best to manage used fuel from nuclear reactors. In this context, irradiation testing of transmutation fuels supports development of the full recycle technology option that would allow repeated recycling of transuranic elements in fast-spectrum reactors for their destruction, reducing the heat load, footprint and radiotoxicity of a geologic waste repository.

Transmutation fuel development activities at INL include design, fabrication, irradiation, and postirradiation examination of metallic and oxide fuels. To date, almost 20 different fuel compositions have been

tested to burnups as high as 30 at.% with constituent compositions of Pu up to 30 wt%, Am up to 12 wt%, Np up to 10 wt%, and Zr between 10 and 60 wt%. In addition to testing various fuel compositions, a number of design innovations are also being investigated for metallic fuels, including: 1) lowering the fuel smear density below the traditional 75% in order to accommodate more swelling, 2) an annular fuel geometry in contact with the cladding to eliminate the need for a sodium bond (with the central hole accommodating fuel swelling), 3) coatings/liners on the cladding inner surface to mitigate fuel-cladding chemical interaction (FCCI) and enable higher temperature operation, 4) alternative major alloying elements to Zr that can stabilize the gamma phase of uranium under irradiation conditions, while increasing fuel alloy melting temperatures or offering other fuel performance benefits, and 5) minor alloy additions to immobilize lanthanide fission products inside the fuel matrix and prevent their transport to the cladding where they make up the primary species responsible for FCCI.

While access to a fast-spectrum test reactor is essential for this research, recent technology readiness guidelines (Carmack et al., 2017) indicate that proof-of-concept irradiation testing can be performed

* Corresponding author.

E-mail address: pavel.medvedev@inl.gov (P. Medvedev).

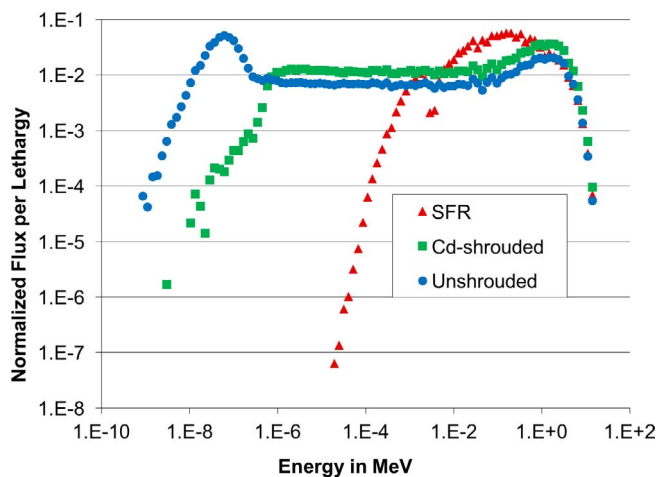


Fig. 1. Impact of Cd shroud on ATR neutron spectrum shown in comparison with unaltered ATR neutron spectrum and SFR neutron spectrum.

using test reactors where irradiation conditions are not fully prototypic of the intended reactor application. As the scope of the proof-of-concept irradiation testing is limited to fuel candidate selection (Crawford et al., 2007), budget, schedule, and practicality considerations led to utilization of the Advanced Test Reactor (ATR) for the proof-of-concept irradiation testing of transmutation fuels. Despite being a thermal spectrum test reactor, ATR is attractive for transmutation fuel testing due to its relatively high fast flux, the ability to tailor the fast-to-thermal flux ratios, and the ability to operate at varying power levels in different regions of the core. Neutron flux tailoring has been used extensively in the ATR using fixed and removable shrouds (Grover, 2006). For the purposes of irradiation testing of transmutation fuels, removable cadmium shrouds of 0.114 cm thickness are used to filter out > 97% of the thermal flux seen by the test articles. Results and methods of the underlying neutron physics analysis describing how ATR neutron flux is tailored for irradiation testing of the transmutation fuels are described elsewhere (Chang and Ambrosek, 2005; Chang, 2011). The resulting neutron spectrum is shown in Fig. 1 in comparison with unaltered ATR neutron spectrum and typical Sodium-cooled Fast Reactor (SFR) neutron spectrum. Other precedent for the use of cadmium to filter thermal neutron flux for the purposes of fast reactor fuel testing dates back to 1960s when cadmium screens were used in BR2 testing of the liquid-metal fast breeder reactor fuel pins under transient operating conditions (Raedt et al., 2000).

From the perspective of testing fuel performance, the principal difference between irradiation of transmutation fuels in fast versus thermal neutron spectra is in how the fission power density is distributed radially in a fuel pellet. Under fully prototypic irradiation conditions achieved in a fast reactor, the power density distribution in the fuel pellet is uniform and independent of the fuel pellet radius since the mean free path of fast neutrons is much larger than the dimensions of a fuel pellet. In a neutron spectrum of a thermal reactor such as ATR, the radial power density distribution in a fuel pellet exhibits significant peaking at the fuel pellet periphery, and a corresponding depression in the central region of the fuel pellet, due to self-shielding, which is attributed to the presence of resonance cross sections of uranium and plutonium isotopes at low neutron energies. Use of cadmium shrouds results in the capture of a vast majority of thermal neutrons by cadmium, greatly reducing self-shielding. Detailed comparison analyses between the cadmium shroud hardened neutron spectrum in the ATR and in a typical fast reactor has been reported by Chang and Ambrosek (2005) and Chang (2011). It was found, that transmutation fuels irradiated in a fast reactor would exhibit a radial peaking factor of 1.01, irradiated in an unshrouded configuration in the ATR – 2.34, and irradiated in an Cd-shrouded configuration in the ATR – 1.23. While

characterization of the radial peaking factors for the neutron spectra of interest illustrates the efficiency of the cadmium shrouds, the relevancy of the irradiation experiments needs to be further demonstrated by comparing fuel temperatures achieved during ATR cadmium-shrouded irradiations of transmutation fuels versus fuel temperatures expected in a fast reactor.

The goal of this paper is to utilize radial power density distributions to evaluate corresponding radial temperature distributions in the cadmium-shrouded irradiation vehicles used for irradiation testing of transmutation fuels and perform comparisons to irradiations in a fast reactor. Formation of the columnar grain region in fast reactor oxide fuels can be easily quantified during postirradiation examination (PIE), allowing determination of the fuel temperature on the outer boundary of the columnar grain region. Therefore, modeling of the columnar grain region formation and comparison of model results to the PIE is an established practice to validate thermal analyses of fast reactor oxide fuel. This is an important step in establishing the relevancy of fuel performance data generated for fast reactor fuels using ATR cadmium-shrouded experiments.

2. Experiment description

For the purpose of comparing calculated columnar grain region dimensions with measurements, the AFC-2C fast reactor mixed oxide (MOX) fuel irradiation experiment has been chosen. Experiment hardware and operating conditions are presented in this section. The rodlet assembly is designed as a miniature length, fast reactor fuel rod. The rodlet assembly consists of the oxide fuel column, stainless steel Type 421 (HT9) cladding, and a fission gas plenum. A stainless steel capsule assembly contains a vertical stack of six rodlets. The capsule and rodlet radial dimensions are shown in Fig. 2. The annular gap between the fuel column and rodlet inner diameter is initially filled by a pure helium bond that is sized to accommodate fuel swelling during irradiation. The annular, helium-filled gap between the rodlet outer diameter and capsule inner diameter is designed to provide the thermal resistance necessary to achieve the target irradiation temperature of the fuel specimen.

2.1. Rodlet assembly

Fig. 3 shows the fuel rodlet assembly axial dimensions. Table 1 shows the materials used in constructing the rodlets along with their

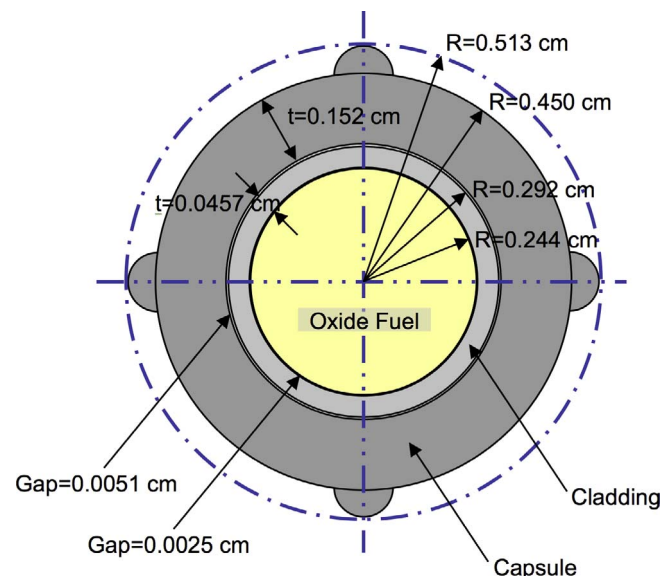


Fig. 2. Nominal radial dimensions of fuel rodlet and capsule assembly for oxide fuel experiments in ATR.

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