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Changing the World's Energy Future

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The Fission Accelerated Steady State Test (FAST) – A Revised Capsule Design for the Accelerated Testing of Advanced Reactor Fuels

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INTRODUCTION

Since 2003, the Idaho National Laboratory (INL) has successfully tested various metallic, oxide, and nitride fuels for fast reactor applications in the thermal neutron spectrum Advanced Test Reactor (ATR) [1]. The testing of fast reactor fuels in ATR is challenging given the thermal neutron spectrum and because ATR coolant is far below coolant temperatures prototypic of liquid metal-cooled fast reactors. The traditional experiment design used for these tests also presents additional challenges given its response under irradiation is highly sensitive to fabrication tolerances. Maintaining prototypic temperatures for fast reactor fuel in a light water reactor testing environment has been achieved using very small gas gaps that must be precisely controlled. Fabrication tolerances of these gas gaps regularly drive up the costs associated with fuel development and testing.

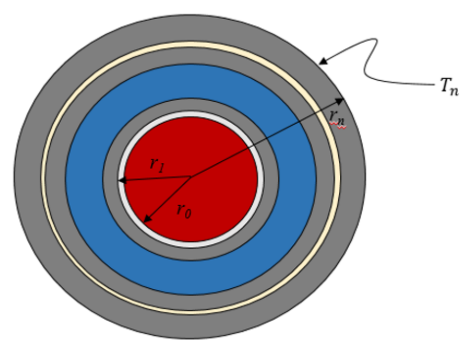
In an effort to improve the development and testing of novel nuclear fuels, a detailed evaluation on reducing the size of fuel specimens for irradiation has been performed. Theoretically speaking, reducing the fuels diameter and incorporating an inner capsule while maintaining the original outside diameter of the experiment significantly reduces the experiment's sensitivity to fabrication tolerances and drastically reduces the time necessary to reach a desired fuel burnup; both of which improve the cost and schedule associated with the fuel testing.

Simulations of this reduced diameter, double encapsulated concept have been performed using Abaqus, MCNP, and ORIGEN. The results of the Abaqus simulations are beyond the scope of this summary and will not be presented herein. For results of these Abaqus simulations, see Beausoleil [2].

The models were tested using various parameters and design deviations which were then compared to the original Advanced Fuels Campaign (AFC) designs. The results of these simulations confirmed that the introduction of a second capsule coupled to a reduction in the fuel pin diameter significantly reduces the experiment's sensitivity to fabrication tolerances and reduces the time to reach a desired burnup by nearly an order of magnitude. The original AFC design requires approximately 11.7 years to reach 30 Atom % Heavy Metal (at% HM) burnup. A burnup of 30 at% HM can be reached in approximately 2.5 to 3.5 years (irradiation position dependent) with a fuel diameter ½ (~0.097 in. diameter) that of the original AFC design. An added benefit to the fuel pin diameter reduction is that more specimens can be irradiated in any given cycle. Essentially, significantly more specimens can be tested and incur more burnup over a given operating cycle relative to the AFC design. This allows the development of a large test matrix of samples with rapid throughput in ATR.

DESIGN PREMISE

The general approach in the revised design is documented in [2] and is to reduce fuel diameter so that power density can be increased while maintaining prototypical temperatures in the fuel and cladding. Figure 1 depicts a schematic of fuel surrounded by concentric cylinders comprised of materials potentially with different thermal conductivities. Also shown are solutions for temperatures at the interfaces between each of the cylinders and the solution for the peak central temperature of the fuel. A key result from the analytic solutions is if the fuel power is characterized in terms of the linear heat generation rate and if the geometry is scaled uniformly, the temperatures reached in all of the concentric cylinders remain unchanged. This simple observation has rather profound implications for the design of experiments to test cylindrical fuel rods in ATR [2].



$$T_i = T_n + \frac{Qr_0^2}{2}\sum_{j=i+1}^{n}\frac{1}{\kappa_j}ln\left(\frac{r_{j-1}}{r_j}\right)$$

$$T_i = T_n + LHGR\sum_{j=i+1}^{n}\frac{1}{2\pi\kappa_j}ln\left(\frac{r_{j-1}}{r_j}\right)$$

$$PICT = T_0 = T_n + LHGR\sum_{j=1}^{n}\frac{1}{2\pi\kappa_j}ln\left(\frac{r_{j-1}}{r_j}\right)$$

$$PCT = PICT + \frac{LHGR}{4\pi\kappa_0}$$

Fig. 1. Fuel schematic with thermal analytical solutions [2].

Given the above expression, scaling a design down by a factor of three while maintaining the same linear heat generation rate results in the same temperature distribution within the fuel and cladding, yet with a factor of nine increase in fuel power density. Because the rate at which fuel burnup accumulates is directly proportional to fuel power density, reducing fuel dimensions by a factor of three could reduce the irradiation time to reach a desired burnup by a factor of three [2]. The idea of increasing power density to reduce irradiation times is not new. In 1961, Blake [3] reported results from experiments where fast reactor fuel pin diameters were reduced to 0.122", which in turn allowed for power densities that achieved 1 atom % burnup for every 17 days of irradiation time.

The problem that arises in simply scaling the current design to smaller dimensions is that the tolerance issues associated with helium-filled gas gap would only be exacerbated. This concern led to the double-encapsulated design shown in Figure 2.

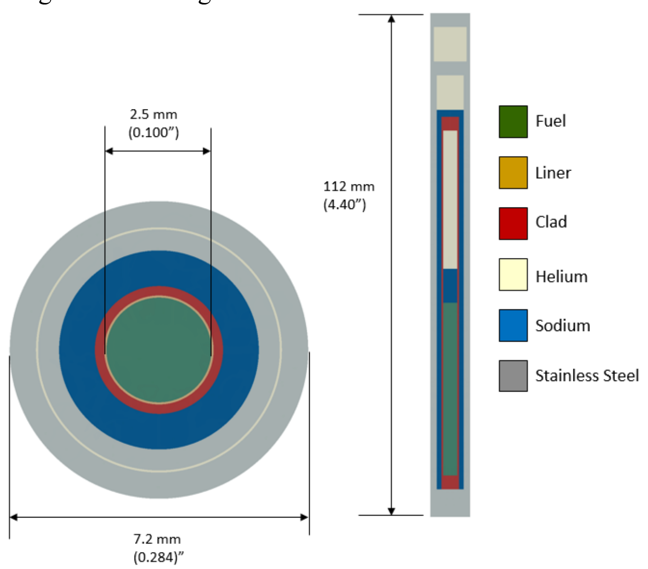


Fig. 2. Simplified double encapsulated reduced diameter design [2].

The helium-filled gap is placed between an inner and outer capsule and liquid sodium is used to thermally bond the inner capsule to the rodlet. The helium gap can be moved toward the periphery of the capsule volume to reduce the heat flux and, everything else being equal, allow for an increase in gap dimensions because the radial heat flux is inversely proportional to the radius. The larger the gap, in turn, the less sensitive the experiment will be to fabrication tolerances; however, analyses have shown that the double-encapsulated design is less sensitive to eccentricities in position of the rodlet and/or inner capsule so that a helium-filled gap on the order of 50 microns (0.002") appears to be acceptable [2].

NUCLEAR ANALYSES

A range of parametric nuclear analyses have been performed to support a preliminary Fission Accelerated Steady State Test (FAST) design. The analyses documented herein were performed utilizing a detailed 3D full-core model of ATR in MCNP, with a capsule model very similar to that depicted in Figure 2. The inner and outer capsules were modeled as 316 stainless steel, the cladding was modeled as HT-9, and the fuel was modeled as U-10Zr with a 75% smear density. A single, double encapsulated and fueled rodlet will be simply called a FAST capsule throughout this summary.

In addition to nine flux traps, ATR contains 68 experiment positions located within the core's beryllium reflector and neck shim housing. Of these positions, the small-I position, I-24, and outer-A position, A-12, were selected given each test position's current availability and the ability to meet LHGRs and temperatures prototypic of fast reactor fuel. The locations of these test positions in ATR are shown in Figure 3.

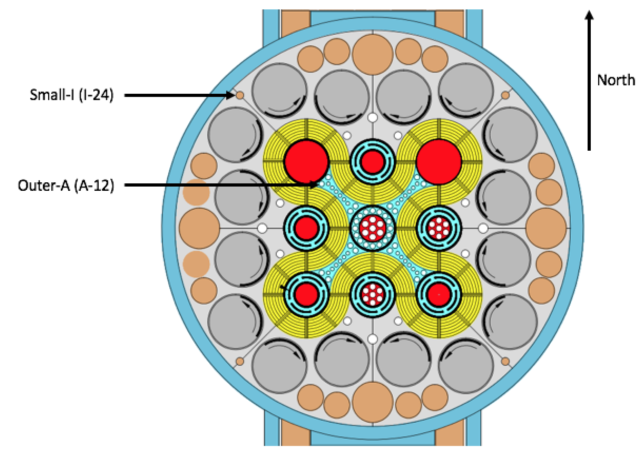


Fig. 3. ATR core cross-section with I-24 and A-12 identified.

The FAST capsules in A-12 were modeled within a heterogenous Aluminum and Cadmium basket assembly. Basket assemblies are used in ATR to ensure each capsule stays in position during irradiation. The FAST capsules irradiated in I-24 were modeled in a basket assembly comprised only of Aluminum (i.e., a Cadmium shroud was not used for the FAST capsules irradiated in I-24). Although there will be substantial differences in the neutron energy spectrum within each irradiation position, the intent is to compare and contrast the behavior of FAST capsules irradiated in I-24 to not only FAST capsules irradiated in the Cadmium shrouded A-12 position, but also to AFC capsules irradiated in Outer-A positions shroud with Cadmium basket assemblies.

The active fuel length of ATR is 48 inches, which allows the loading 8 capsules per irradiation position. The current and preliminary FAST capsule design is ~ 5.5 inches in overall length. The small-I positions are larger than the outer-A positions, allowing for the design and use of a basket that will hold three individual stacks of 8 FAST capsules, for a total of 24 FAST capsules in I-24. Figure 4 is a simplified rendition of the FAST basket assembly in the I-24 position. Note the three FAST irradiation position (Channels 1 through 3) in the I-24 basket and the three smaller diameter positions available for flux monitors.

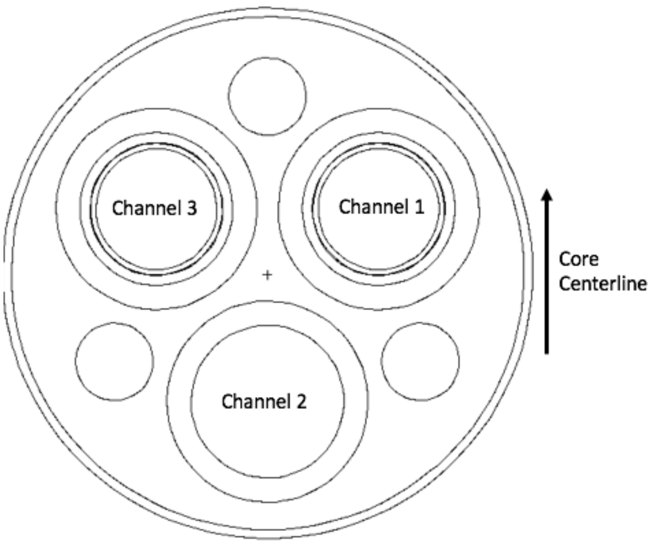


Fig. 4. I-24 FAST basket assembly

RESULTS

MCNP [4] simulations coupled to ORIGEN2 [5] activation and depletion calculations were used to generate all results documented in this section. Note that the FAST fuels described were modeled with a diameter of 0.097 inches (~ ½ that of typical AFC fuel)

Fuel Linear Heat Generation Rates (LHGR) ranging from 300-350 W/cm in the I-24 position, and 200-250 W/cm in the A-12 position were targeted to meet temperatures prototypic of fast reactor fuels. The only fuel type analyzed was U-10Zr, with a 75 % smear density. Table 1 and Table 2 list fuel enrichment and calculated LHGRs at beginning of life for the capsules irradiated in Channel 1 of I-24, and for the FAST capsules in A-12, respectively. Aside for the two capsules at the top and bottom of the stack-up in each irradiation position, the target LHGRs are met.

TABLE 1. FAST LHGRs in I-24 (No Cadmium Shroud)

FAST Capsule	wt/o 235U	LHGR (W/cm)	Distance From Core Centerline (in.)
Capsule 8	93.0	155	20.1
Capsule 7	93.0	274	14.4
Capsule 6	60.0	320	8.6
Capsule 5	49.5	331	2.9
Capsule 4	49.5	331	-2.9
Capsule 3	60.0	320	-8.6
Capsule 2	93.0	283	-14.4
Capsule 1	93.0	158	-20.1

TABLE 2. FAST LHGRs in A-12 (With Cadmium Shroud)

FAST Capsule	wt/o 235U	LHGR (W/cm)	Distance From Core Centerline (in.)
Capsule 8	93.0	113	19.4
Capsule 7	93.0	168	13.9
Capsule 6	93.0	202	8.3
Capsule 5	93.0	217	2.8
Capsule 4	93.0	218	-2.8
Capsule 3	93.0	202	-8.3
Capsule 2	93.0	157	-13.9
Capsule 1	93.0	93	-19.4

A comparison of the burnup rate for the FAST capsules analyzed in the I-24 and A-12 positions are provided in Tables 3 and 4, respectively. The depletions performed were six, 60 full power day ATR cycles with 30 day outages. ATR cycles typically operate for 55-60 days at full power with 28 day outages.

TABLE 3. Burnup Accumulation in Channel 1, I-24 (No Cadmium Shroud)

FAST Capsule	Burnup After 360 Days (at % HM)	Averaged Burnup per 60 Day Cycle (at % HM)
Capsule 8	12.3	2.1
Capsule 7	20.9	3.5
Capsule 6	22.6	3.8
Capsule 5	22.5	3.8
Capsule 4	22.6	3.8
Capsule 3	22.8	3.8
Capsule 2	21.6	3.6
Capsule 1	12.4	2.1

TABLE 4. Burnup Accumulation in A-12 (With Cadmium Shroud)

FAST Capsule	Burnup After 360 Days (at % HM)	Averaged Burnup per 60 Day Cycle (at % HM)
Capsule 8	7.3	1.2
Capsule 7	11.9	2.0
Capsule 6	15.1	2.5
Capsule 5	16.2	2.7
Capsule 4	16.0	2.7
Capsule 3	15.1	2.5
Capsule 2	12.8	2.1
Capsule 1	8.8	1.5

The standard diameter AFC capsules operating at 350 W/cm accumulate approximately 0.76 at/% HM burnup per 60 full power day cycle [2]. Assuming ATR operates for 200 full power days per year, the standard diameter AFC capsules require ~11.7 years of irradiation to reach 30 at% HM burnup. The half diameter I-24 FAST capsules operating at 330 W/cm at beginning of life require ~2.4 years, and the lower power half diameter FAST capsules irradiated within the Cadmium basket assembly in the A-12 position require 3.3 years. Although the calculated power densities are not shown, the results confirmed substantial increases in power density, which is apparent given the increased rate of burnup accumulation. Blake [3] reported 1.0 at % for every 17 days of irradiation time for 0.122" diameter fuel pins. The results from Blake's work provides confirmation of these results.

Figure 5 provides a comparison of the calculated radial power profiles (the ratio of local power to the average power within a fuel specimen). The results from four simulations are shown: standard (nominal) diameter fuel with and without a Cadmium shroud, and half diameter fuel with and without a Cadmium shroud. As expected, both the use of a Cadmium basket and an increase in fuel power density result in a more uniform radial power distribution through the fuel. Given that reducing fuel diameter by a factor of two without a cadmium shroud produces a similar radial power distribution as the full diameter case with Cadmium, it is apparent that acceptable radial power distributions are achieved with a reduction of fuel diameter.

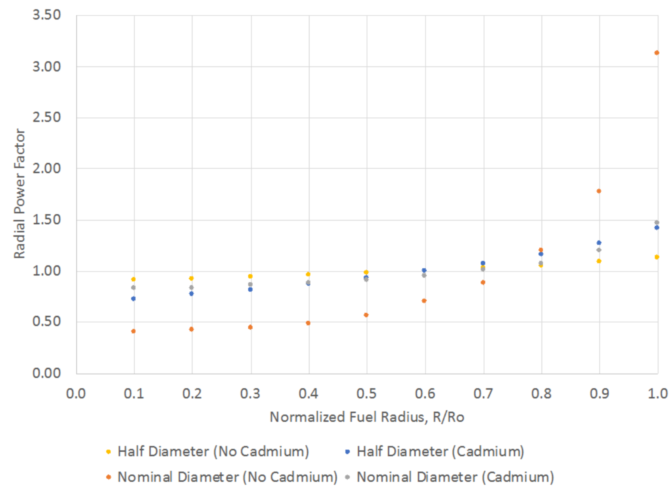


Fig. 5. Radial Power Comparison

CONCLUSIONS

The FAST capsule design exploits reduced diameter fueled experiments to increase fuel power density and markedly increase the rate of burnup accumulation relative to standard diameter fuel types irradiated within a Cadmium shroud. The analyses documented herein demonstrated that the time to reach a burnup of 30 at% HM can be reduced from ~11.7 years using the standard diameter design to roughly 2.5 to 3.5 years. This reduced diameter design also discards the need for a Cadmium basket to improve the radial power distribution within the fuel, and allows for the use of small-I irradiation assemblies that can irradiate up to 24 FAST capsules per small-I position. This increase in throughput coupled to a significant reduction in time to reach high fuel burnups allows for expedited testing of novel fuels and materials for fast neutron reactors.

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