Initial Design of High-Temperature MiniFuel Irradiation Experiments in the High Flux Isotope Reactor Removable Beryllium Region

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INTRODUCTION

The qualification of novel nuclear fuel concepts requires irradiation experiments that have conventionally been performed in an integral testing format. In this format, the experimental conditions are designed to mimic the expected operational conditions the fuel will experience in its prototypic geometric form. A key challenge associated with integral fuel testing is the number of independent variables related to fuel performance and safety that often vary simultaneously, resulting in large test matrices and large uncertainties in experimental data and the fuel performance models that depend on that data. Irradiation testing of miniature fuel specimens (or *MiniFuel)* has been conducted in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) with the purpose of isolating the independent variables that impact fuel performance and safety characteristics, and to rapidly accumulate separate effects irradiation data that can be used to qualify nuclear fuels and reduce uncertainties in fundamental fuel performance models [1].

The small size of the MiniFuel specimens causes most of the heat within the experimental capsules to come from gamma heating in the capsule materials rather than from fission heating in the fuel, allowing for the fuel temperature to be well-controlled with minimal temperature gradients. To date, all MiniFuel experiments have been conducted in HFIR's vertical experimental facility (VXF) positions and have generally targeted average fuel temperatures in the vicinity of 450–550°C. The purpose of this study is to modify the MiniFuel irradiation vehicle for testing in HFIR's removable beryllium (RB) positions, which is closer to the reactor core and will enable high-temperature (i.e., >1000°C) irradiation tests that accumulate burnup more rapidly than experiments in VXF positions. Reactor physics, thermal hydraulics, and finite element thermal analysis methods are used to design an initial MiniFuel RB experiment to meet a set of demonstration design goals.

METHODOLOGY

Geometric Design

The primary subassemblies of the RB MiniFuel irradiation vehicle are the basket, the targets, and the individual sub-capsules (Fig. 1). The basket is an aluminum holder assembly with five radial holder positions that can each hold three axially stacked targets (15 total targets). Each target contains six sub-capsules, which consist of a holder and cap, filler, specimen cup, fuel specimen, silicon carbide (SiC) passive thermometry, and grafoil insulators. The holder, filler, cup, and cap are all made of molybdenum, a refractory metal with high density, melting point, thermal conductivity, and minimal chemical interaction with most uranium-based fuels. Sub-capsules are separated by thimbles that keep them centered within the stainless-steel target housing. Springs are located on each end of the stack of sub-capsules to insulate the sub-capsules and allow for thermal expansion. The target is sealed with a fill gas consisting of pure helium or a helium and argon mixture. Fig. 1 shows a diagram of the irradiation vehicle assembly and a breakdown of the individual components. Radial-Axial-Sub-capsule (RAS) position identifiers are included in the diagram. For this study, the fuel specimens are 3-mm diameter, natural UO_2 disks with variable thickness.

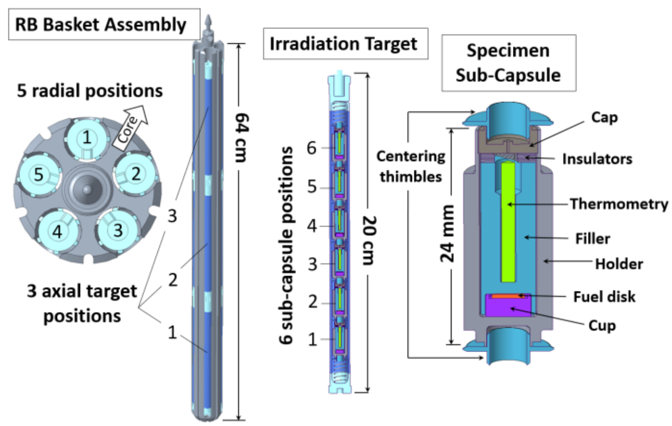

Fig. 1. Irradiation vehicle assembly diagram.

Neutronics Analysis Methods

Neutron and photon heating rates in all irradiation vehicle components were calculated using MCNP5 [2], the ADVANTG variance reduction tool [3], and the ORIGEN depletion module of the SCALE software package [4], which are externally coupled together using an ORNL-developed wrapper called HFIRCON. A previously-developed model of HFIR cycle 400 (April 27, 2004 – May 21, 2004) [5] was modified to include the RB irradiation vehicle assembly and was used to calculate heat generation rates (HGRs) from neutron and photon sources in all materials, fuel burnup, and fuel fission rates as a function of irradiation time. At each timestep, several MCNP calculations are performed to

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calculate prompt heating rates, and neutron and photon flux spectra are passed between MCNP and ORIGEN to account for local α and β decay heating. ORIGEN calculates isotopic compositions and passes that information into MCNP to update the model to allow for multi-cycle calculations. Four irradiation cycles were simulated that assumed 26-day operational periods followed by 25-day downtimes between cycles.

Thermal Hydraulic Analysis Methods

A model of the MiniFuel irradiation vehicle assembly seated in the aluminum liner of one of HFIR's RB testing positions was developed in RELAP5-3D [6] to determine the coolant mass flow rate through the experiment and to determine the heat transfer coefficient (HTC) between the target assemblies and the coolant. The mass flow rate through the assembly was calculated to ensure the result was above the safety limit for one of HFIR's irradiation positions (18 GPM, or approximately 1.12 kg/s), and the HTC was calculated and later used in the finite element (FE) thermal analyses to define the convective boundary condition. RELAP calculations were performed using the nominal geometric dimensions, and two additional cases were run in which the dimensions were varied within their geometric tolerances to give the minimum and maximum possible cross sectional flow area. This approach allowed for the range of possible mass flow rates and HTCs to be determined.

A nodalization diagram of the RELAP model is shown in Fig. 2 and consists of an inlet time-dependent volume flowing into the open aluminum liner above the irradiation vehicle, through a multi-junction that redirects some flow into the five holder slots around the targets and some flow around the basket assembly in between the liner. Lateral cross flow is modeled so that coolant can flow between the two channels. Coolant from both channels meets at another multi-junction and then flows through the bottom of the basket assembly and liner and into a sink volume. Three heat structures are included in the model: one which represents the RB liner, one which represents the basket, and one which represents all the targets lumped together. By modeling the targets as their own heat structure and separating the flow channel through the five radial holder positions from flow around the basket, the HTC between the targets and the local coolant could be isolated.

The total power in the lumped targets heat structure was the sum of the heat calculated in each of the 15 targets using HFIRCON. HGRs in the aluminum liner and basket, and direct heating in the coolant were assumed to be equivalent to values calculated for a previous HFIR RB experiment. Although the capsule design in the reference calculation differed from the MiniFuel basket design, HGRs in a given material have been shown to be relatively insensitive to the geometry when normalized to mass.

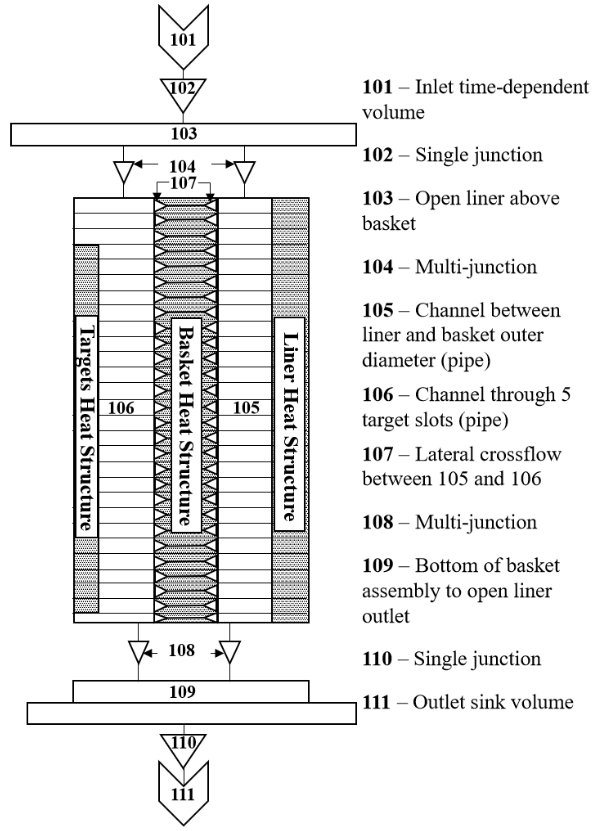

Fig. 2. Nodalization of RELAP5 MiniFuel basket model.

FE Thermal Analysis Methods

Three-dimensional FE analyses were performed using an adapted quarter-symmetry model of a single target developed in the ANSYS software package and used in previous MiniFuel analyses [1, 7]. A convective boundary condition using a bulk coolant temperature of 58°C and the average HTC determined using the RELAP5-3D model acted as the ultimate heat sink. A custom macro was used to read the HGRs calculated using HFIRCON and assign internal heat generation rates to each component in the target subassembly. The thermal conductivity of each material in the model was temperature dependent, and the thermal conductivity of the fuel was also dependent on burnup. A mesh sensitivity was performed in which the element size in the fuel and in all of the nonfuel components were reduced until temperatures changed by less than 1°C. It was shown that node lengths of 0.5 mm in nonfuel components and 0.1 mm in the fuel specimens were sufficient for mesh convergence. Fig. 3 shows the quarter symmetry ANSYS target model.

Fig. 3. Quarter-symmetry ANSYS target model.

A preliminary test matrix (Table I) was developed for a planned initial RB MiniFuel experiment targeting improved understanding of high-temperature fission gas release in UO_2 [8]. Several parameters in the ANSYS model were iterated on until the design goals were met. The primary parameters that could be changed in the model were the gas gap thickness between each sub-capsule and the target housing inner surface, the thickness of the fuel disk, and the target fill gas composition. In addition to the design goals listed in Table I, it was desirable to minimize the spatial and temporal ΔT in each fuel specimen.

TABLE I. Preliminary design goals

Goal Average Fuel Temperature [°C]	Goal Burnup [MWd/kg-U]	Number of specimens [-]
1000	40	2
1000	80	2
1100	40	2
1100	80	2
1300	40	2
1300	80	2

RESULTS

The reactor physics calculations performed using HFIRCON showed that the HGRs in the fuel and nonfuel components varied significantly as a function of target position in the basket, but HGRs in a given position remained relatively constant over time after the initial buildup of fissile plutonium isotopes that occurs over the first cycle of irradiation. Large variations in HGRs also occurred over the first day of each subsequent irradiation cycle due to the HFIR startup sequence and buildup of fission products in the HFIR fuel. Fig. 4 shows the evolution of the total HGR in all fuel specimens in the three targets in radial position 1 (refer to Fig. 1 for position identifiers) at the beginning, middle, and end of the fourth irradiation cycle. The figure illustrates the spatial and temporal dependence of HGRs.

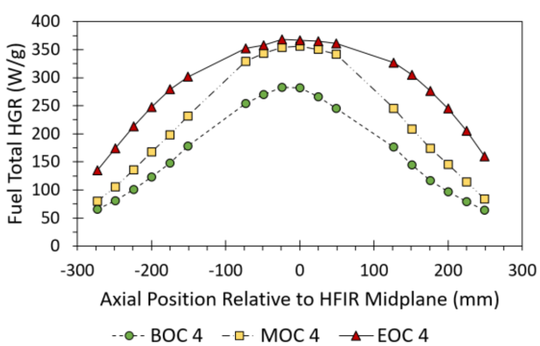

Fig. 4. Axial and temporal dependence of fuel HGRs.

Due to the proximity of the RB position to HFIR's core, the HGRs and burnup accumulation in the fuel is greater than that in the VXF positions. Fig. 5 compares the fuel fission heating and burnup accumulation in the sub-capsule with the greatest HGR from the current study to those from a previous study on VXF MiniFuel experiments using natural UO_2 fuel specimens [1]. In Fig. 5, the RB data was adjusted to have 15-day downtimes between cycles to reflect the same reactor downtime assumed in the previous VXF work. This adjustment has no impact on the burnup since burnup is not accumulated while the reactor is off. The figure shows that the HGR and burnup accumulation is approximately 2.25 times greater in the RB position compared to in the VXF position.

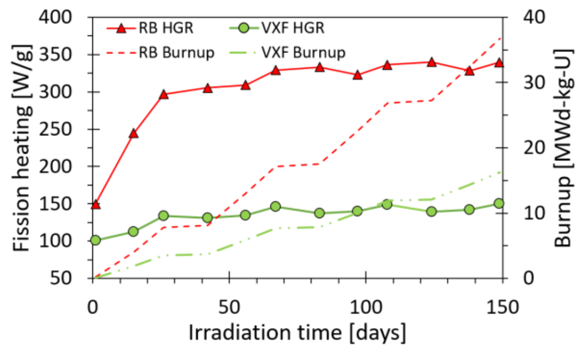

Fig. 5. Comparison of HGR and burnup for natural UO_2 in an RB and VXF position.

Coolant mass flow rate predictions from the RELAP model showed that the minimum, nominal, and maximum flow rates were 1.21 kg/s, 1.36 kg/s, and 1.52 kg/s, respectively. The minimum possible flow rate is above the lowest acceptable flow rate of 1.12 kg/s. The predicted HTCs were relatively insensitive to the variations in the model geometry. The nominal averaged HTC was approximately 22.5 kW/m^2-K, and the minimum and maximum flow area cases caused a variation from the nominal value of +/-0.34 kW/m^2-K. Scoping calculations with the ANSYS model showed that this degree of HTC variation was inconsequential to the fuel temperature, and only the nominal HTC was considered for FE calculations.

Inspection of the fuel burnups predicted with HFIRCON showed that the maximum burnup reached after four irradiation cycles occurred in target position 12, and the six specimens in the target had burnups ranging from 35.8–38.9 MWd/kg-U. Linear extrapolation was used to identify a target that would reach 80 MWd/kg-U, and it was found that the fuel specimens in target 32 would have burnups ranging from 76.1–81.4 MWd/kg-U after 9 irradiation cycles. These two target positions were the focus of the FE thermal analysis. Due to the temporal variation in HGRs, the design parameters in the ANSYS model were iterated on until the average fuel temperature goals listed in Table I were approximately met using a middle of cycle (MOC) burnup and HGR. This value is representative of an overall average HGR when the variations during the first irradiation cycle and during HFIR's startup sequence are ignored. Figs. 6 and 7 show the minimum, average, and maximum fuel temperature of each fuel specimen throughout the four irradiation cycles. In the figures, the markers represent the average temperatures, and the error bars represent the range from the minimum to maximum fuel temperature.

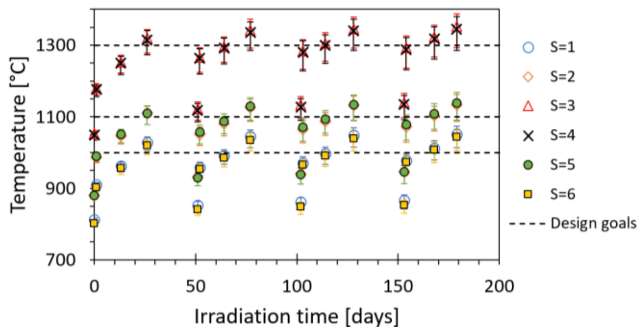

Fig. 6. Fuel temperature predictions in RA position 12 throughout 4 irradiation cycles.

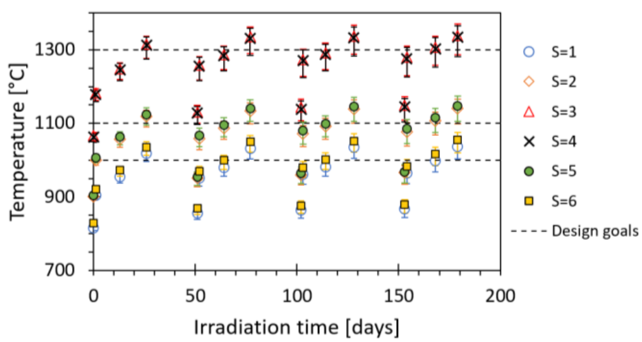

Fig. 7. Fuel temperature predictions in RA position 32 throughout 4 irradiation cycles.

The peak spatial ΔT that occurred in any of the fuel specimens in positions 12 and 32 during the four-cycle irradiation was 94.4°C and occurred in RAS position 124. Figs. 6 and 7 show that the goal average fuel temperatures in Table I could be achieved while maintaining a ΔT below 100°C. Ignoring the temperature variation during reactor downtime and during the startup sequence of each irradiation cycle, the maximum temporal ΔT was 165.6°C and occurred in RAS position 123.

CONCLUSION

An initial design study on high-temperature MiniFuel irradiation experiments in HFIR's RB region has been conducted. Monte Carlo neutronics methods showed that compared to MiniFuel experiments conducted in the VXF region, MiniFuel RB experiments could achieve 2.25 times greater HGRs and burnup accumulation rates in natural UO_2. Extrapolation of burnup values shows that burnups as high as 80 MWd/kg-U could be achieved after 9 irradiation cycles. Thermal hydraulics analysis that accounted for variations in geometry due to dimensional tolerances showed that the minimum possible coolant flow rate through the experiment satisfied the HFIR safety limit, and the HTC between the targets and coolant was insensitive to the geometric variation. Iterative FE analysis methods were used to meet a set of preliminary design goals relevant to high-temperature light water reactor scenarios and physical phenomena. This study showed that fuel temperatures greater than 1000°C are obtainable in the RB region. It was also shown that fuel specimens could differ in average temperature by at least 300°C within the same target while maintaining spatial ΔT values below 100°C. Even greater temperature ranges are likely achievable by utilizing different target positions and combinations of gas gap thickness, fuel thickness, and fill gas composition. Future work will consist of more specific design goals from potential vendors to improve modeling accuracy and accelerate nuclear fuel qualification.

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