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Irradiation Plan for a Mixed Thorium-Uranium Oxide Drop-In Experiment in the Advanced Test Reactor

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ABSTRACT

Clean Core Thorium Energy (CCTE), LLC is committed to the development of alternative nuclear fuels using thorium, primarily for pressurized heavy-water reactors and Canada deuterium uranium (CANDU) reactors. CCTE requested Idaho National Laboratory (INL) support to conduct irradiation testing on various mixed thorium-uranium oxide ($(Th,U)O_2$) fuel samples to help assess fuel performance. Experiment pellets were fabricated at Texas A&M University under direction from CCTE.

Over the next several years, twelve experiment capsules, each containing 18 experimental fuel pellets, are planned to be irradiated in the Inner-A and Outer-A experiment positions of the Advanced Test Reactor (ATR) at INL. Experiment pellets with varying compositions of $(Th,U)O_2$ will be irradiated across a range of burnup targets to assess the potential benefits of the CCTE-designed "Advanced Nuclear Energy for Enriched Life" (ANEEL) fuel. The planned burnup targets are well in excess of the burnup limits for the traditional natural uranium fuel used in CANDU reactors, which means that ANEEL fuel has the potential to dramatically increase fuel utilization and extend operation for those reactors. As the experiment pellets reach their desired burnup levels, they will be removed from the ATR and sent to the INL Hot Fuels Examination Facility for post-irradiation examination.

This paper outlines the irradiation plan for the CCTE-ANEEL-1A experiment including irradiation locations, test train geometry, and a summary of the experiment design analysis. ATR irradiation is expected to begin in Spring 2024, with burnup targets of 20, 40, and 60 GWd/MTU.

Keywords: ATR, CCTE, Thorium, Irradiation Experiment, ANEEL

1. INTRODUCTION

Clean Core Thorium Energy (CCTE), LLC located in Oak Brook, Illinois, is committed to developing alternative nuclear fuels, especially thorium-based fuels for use in the nuclear industry. CCTE has requested Battelle Energy Alliance, LLC (BEA), the U.S. Department of Energy's (DOE) Management and Operating Contractor for Idaho National Laboratory (INL), located in Idaho Falls, Idaho, to execute fuel testing for the advancement of their development objectives.

CCTE requested INL's support to conduct irradiation testing on mixed thorium-uranium oxide ($(Th,U)O_2$) fuel samples to assess fuel performance over a range of burnups at heat generation rates typical of

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pressurized heavy-water reactors (PHWRs) and Canada deuterium uranium (CANDU) reactors. The CCTE-ANEEL-1A irradiation test will be conducted in the Advanced Test Reactor (ATR) at INL as a first step toward development and qualification of the “Advanced Nuclear Energy for Enriched Life” (ANEEL) fuel system designed by CCTE. Because the planned burnup targets for this irradiation test are well in excess of the burnup limits for traditional natural uranium fuel used in CANDU reactors, the CCTE-designed ANEEL fuel has the potential to dramatically increase fuel utilization and extend operation for those reactors. Texas A&M University (TAMU) fabricated the experiment fuel pellets used in the CCTE-ANEEL-1A irradiation test and INL performed the design, analysis, fabrication of experiment hardware, and assembly of the test capsules.

2. IRRADIATION EXPERIMENT DESCRIPTION

To achieve the test objectives, the CCTE-ANEEL-1A experiment will be irradiated in the Inner-A and Outer-A positions on the southern side of the ATR at INL. Southern locations are preferable to northern locations due to the higher powers typically experienced in the southern lobes. A core cross section indicating these locations is shown in Figure 1.

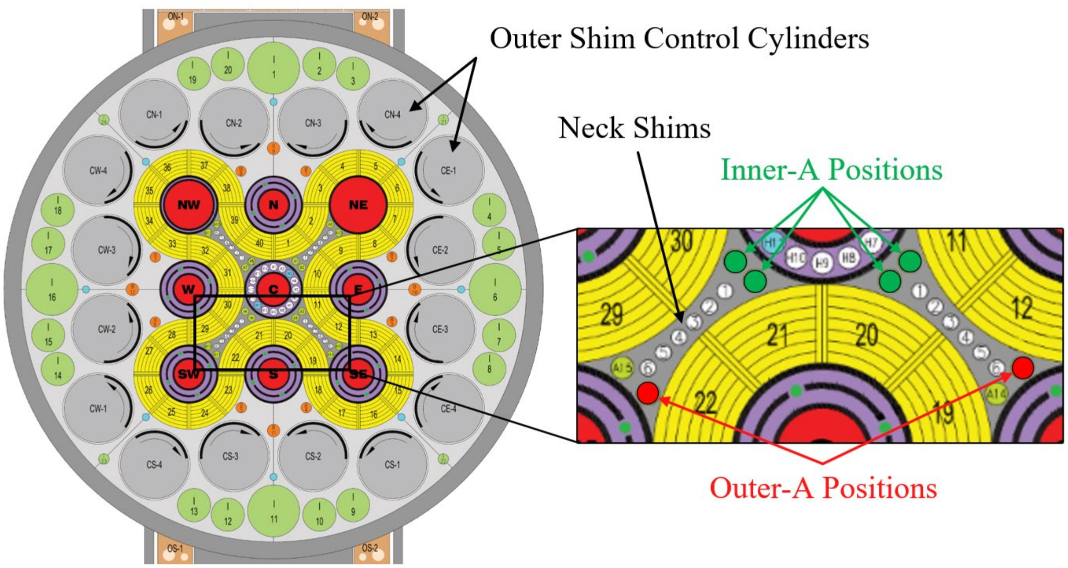

Figure 1. CCTE-ANEEL-1A experiment locations in ATR.

During irradiation, reactor power is controlled via rotation of the Outer Shim Control Cylinders (OSCCs) and insertion/removal of the neck shims. Both control devices are noted in Figure 1. Of particular note to this experiment is that the neck shims typically start an irradiation cycle fully inserted and then are withdrawn from position 6 (outermost) to position 1 (innermost) as the irradiation cycle progresses. Given their proximity to the Inner-A and Outer-A experiment positions, heating rates in those positions are affected by this progressive withdrawal of the neck shims. More discussion on this effect can be found in Section 3.1.3.

Three test trains (CCTE-L, CCTE-M, and CCTE-H) have been constructed with each train containing four test capsules (12 total capsules). Each test train will be irradiated until the burnup targets of 20, 40, and 60 GWd/MTU, respectively, are achieved. The test will begin with all test trains in the Inner-A positions noted in Figure 1. The actual Inner-A positions used will be based on ATR availability. Once sufficient burnup is achieved, the test trains will be moved to the Outer-A positions noted in Figure 1. It is expected that seven ATR cycles will be necessary to reach the highest burnup target of 60 GWd/MTU. A notional irradiation schedule for the CCTE-ANEEL-1A experiment is shown in Figure 2.

	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5	Cycle 6	Cycle 7
CCTE-L	Inner-A	Inner-A					
CCTE-M	Inner-A	Inner-A	Inner-A	Outer-A			
CCTE-H	Inner-A	Inner-A	Inner-A	Outer-A	Outer-A	Outer-A	Outer-A

Figure 2. Planned irradiation schedule for the CCTE-ANEEL-1A experiment.

Following irradiation, the CCTE-ANEEL-1A capsules will be cooled in the ATR canal and shipped to INL's Hot Fuels Examination Facility (HFEF) for Post-Irradiation Examination (PIE). Planned PIE scope includes visual inspection of the experiment capsules, neutron radiography, profilometry, fission gas composition analysis, and optical microscopy. PIE efforts are expected to extend through 2027.

2.1. Test Specimen Description

The CCTE-ANEEL-1A experiment consists of four pellet types with varying ratios of $(Th,U)O_2$. All of the pellets were fabricated by TAMU under the direction of CCTE and shipped to INL for irradiation. Upon receipt of the pellets, INL performed visual, dimensional, and mass evaluations for all pellets to verify reported values by TAMU. Several pellets were also sent for destructive analysis to verify chemical and isotopic compositions. A representative photo of an experiment pellet can be seen in Figure 3. Unlike pellets from commercial manufacturers, surfaces of the pellets are not polished.

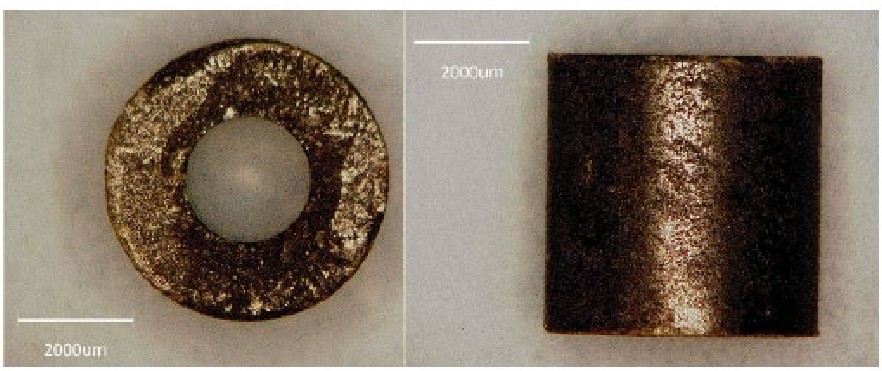

Figure 3. Annular CCTE-ANEEL-1A experiement pellet.

2.2. Test Capsule Description

Each CCTE-ANEEL-1A test capsule consists of a stack of 18 experimental pellets of the same kind with solid ZrO_2 thermal insulating pellets (TIPs) at either end of the stack. The pellet stack is clad in Grade 316H Stainless Steel (SS316H) to form a rodlet. The cladding temperature is expected to be too high for Zircaloy-4 to be used. Each rodlet will be encapsulated in an SS316L outer capsule. A representative cross-section of a test capsule is shown in Figure 4 and a pellet stackup prior to insertion into the rodlet cladding is shown in Figure 5.

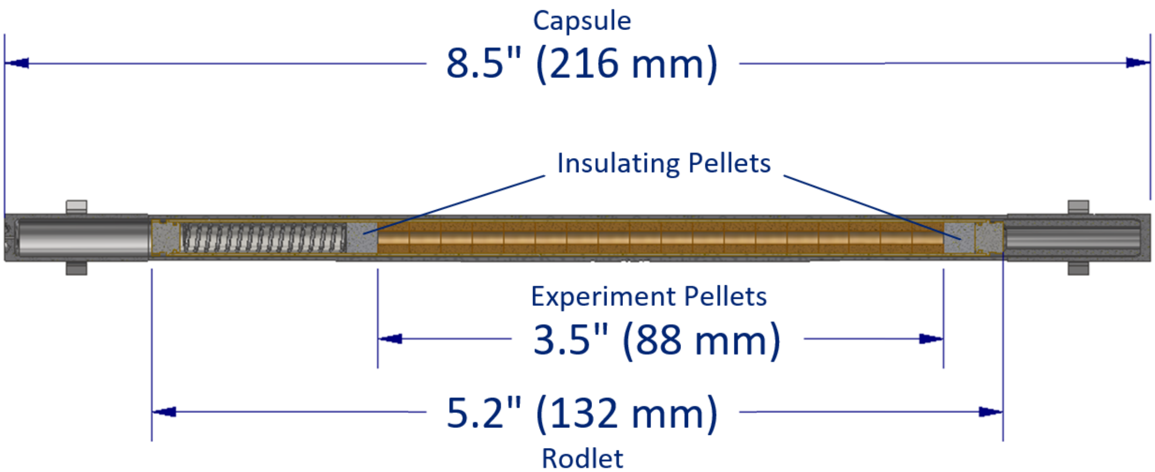

Figure 4. Representative cross-section of the CCTE-ANEEL-1A test capsule.

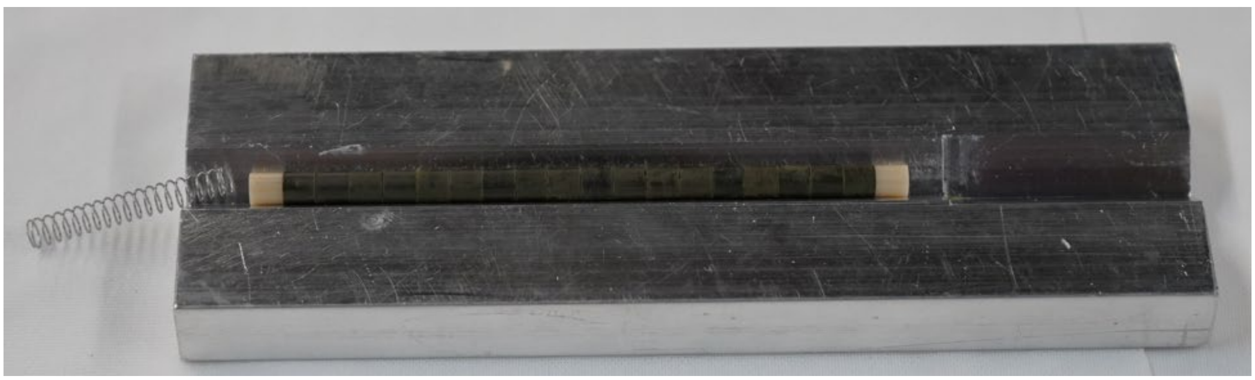

Figure 5. CCTE-ANEEL-1A pellet stack prior to insertion into the rodlet cladding.

2.3. Test Train Description

Each CCTE-ANEEL-1A test train consists of four test capsules, each containing a different pellet type, stacked vertically in a standard ATR irradiation basket. Axial positions of the capsules are selected to achieve a similar heating rate across all capsule types for the majority of irradiation. An example of the test train stackup can be seen in Figure 6. Capsule 4 contains the highest amount of ThO_2 (and subsequently the least amount of UO_2) at beginning-of-life so it is placed closest to the ATR core midplane to maximize the burnup in that capsule. Capsules 1 and 2 contain the least amount of ThO_2 (and subsequently the most amount of UO_2) so they are placed at the outer positions of the stack to reduce beginning-of-life heating.

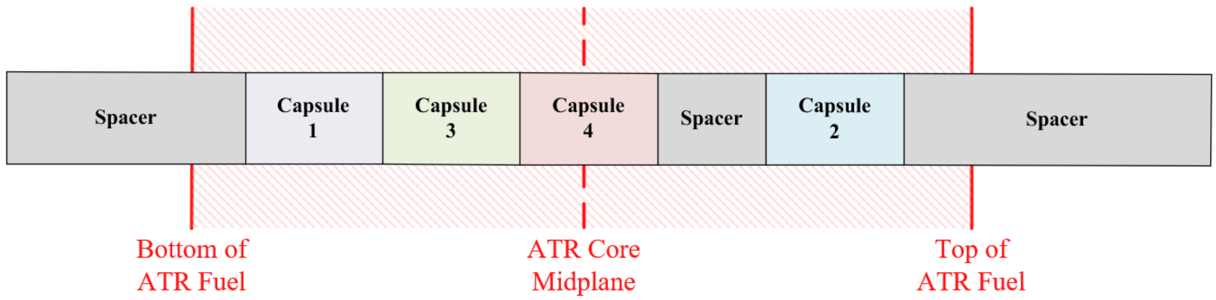

Figure 6. Sample test train stackup for the CCTE-ANEEL-1A experiment.

3. DESIGN ANALYSIS SUMMARY

This section presents a summary of the design analysis for the CCTE-ANEEL-1A irradiation test based on the nominal irradiation schedule in Figure 2. The design analysis consists of nuclear, thermal, and structural analyses. Safety analyses for the experiment in the ATR were also performed, but they are not included in this summary as they are not relevant to the performance of the experiment.

3.1. Nuclear Analysis

3.1.1. Sample burnup

One of the primary objectives of the CCTE-ANEEL-1A experiment is to show that the CCTE-designed fuel is able to go to much higher levels of burnup than typically seen in traditional natural uranium fuels. As such, meeting the burnup objectives is paramount. Table I shows the average calculated level of burnup for each of three test trains based on the irradiation schedule in Figure 2. Each irradiation cycle is assumed to be 60 days long with an average ATR total core power of 108 MW, which is representative of a typical ATR cycle. The Metric ton of uranium (MTU) value used in the burnup calculation is the sum of the nominal fresh masses of uranium and thorium.

Table I. Test train average burnup for the CCTE-ANEEL-1A experiment.

	Test Train Average Burnup (GWd/MTU)						
	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5	Cycle 6	Cycle 7
CCTE-L	10	20	-	-	-	-	-
CCTE-M	11	20	29	38	-	-	-
CCTE-H	10	20	28	37	45	53	58

3.1.2. Fissile mass inventory

One of the main advantages of using thorium in nuclear fuel is that ^{233}U, a fissile isotope, is created from absorption and beta-decay from ^{232}Th. This breeding of fissile material extends the potential lifetime of the nuclear fuel. Figure 7 shows the total calculated fissile material ($^{235}U + ^{233}U$) for the different capsule types in the CCTE-ANEEL-1A experiment as a function of burnup. Capsule 1 has the least amount of ThO_2 (pure UO_2) and thus experiences a monotonic decrease in fissile inventory. Capsule 4 contains the most amount of ThO_2 and therefore sees a much flatter fissile mass profile. Note that some labels have been intentionally left off of Figure 7 to protect proprietary information.

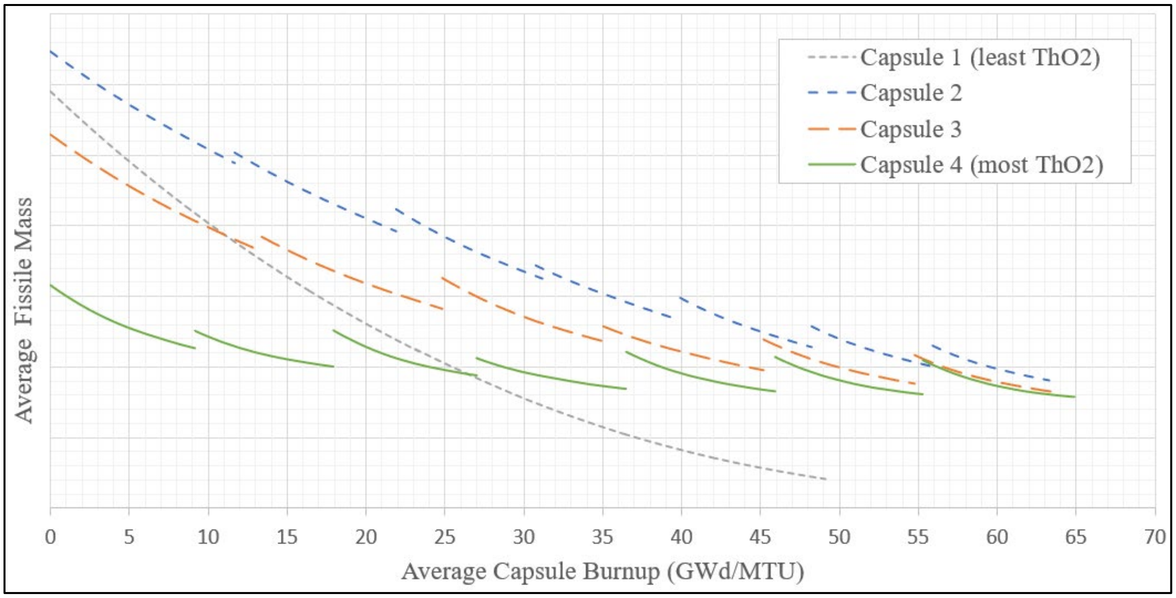

Figure 7. Average capsule fissile mass ($^{235}U + ^{233}U$) as a function of burnup.

To further illustrate the effect of using thorium in nuclear fuel, Figure 8 shows the average percentage of the power in a given capsule that is generated from fissions of ^{233}U instead of fissions of ^{235}U. Capsule 1 is excluded from the figure because it does not contain any thorium. By the end of the irradiation, at least 70% of the fissions in any given capsule are coming from ^{233}U instead of ^{235}U, which indicates good conversion of the thorium.

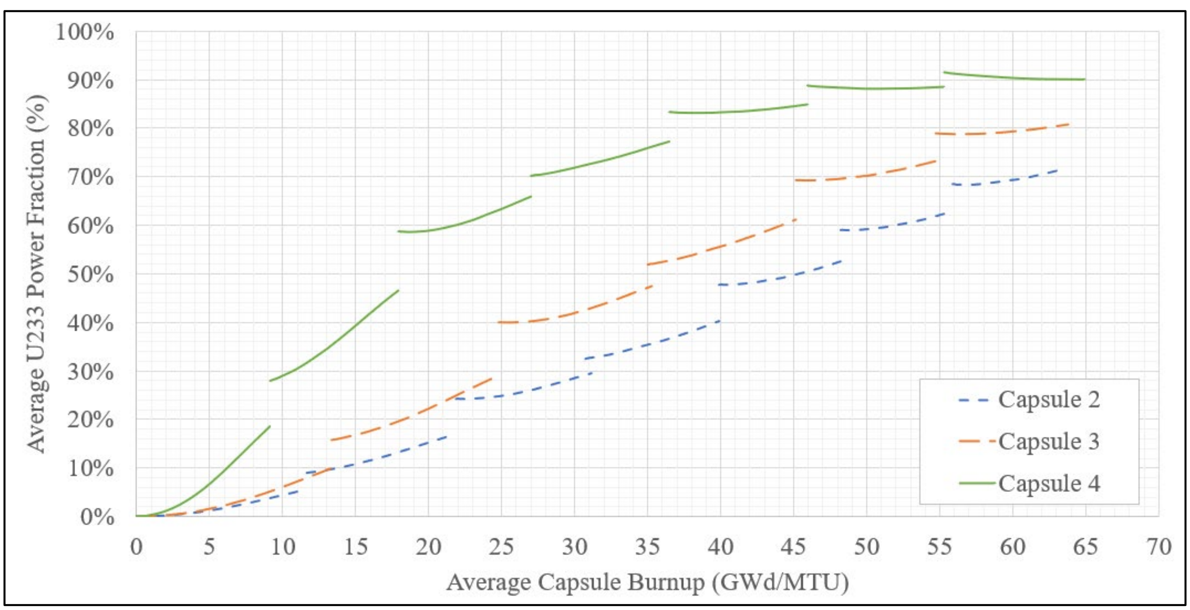

Figure 8. Average capsule ^{233}U power fraction as a function of burnup.

3.1.3. Heat generation rate

Figure 9 shows the average calculated heating rate for each capsule type as a function of burnup. By design, all of the capsules experience a similar heating rate throughout the bulk of the irradiation. Each line segment in Figure 9 represents an ATR cycle. Note that some labels have been intentionally left off of Figure 9 to protect proprietary information.

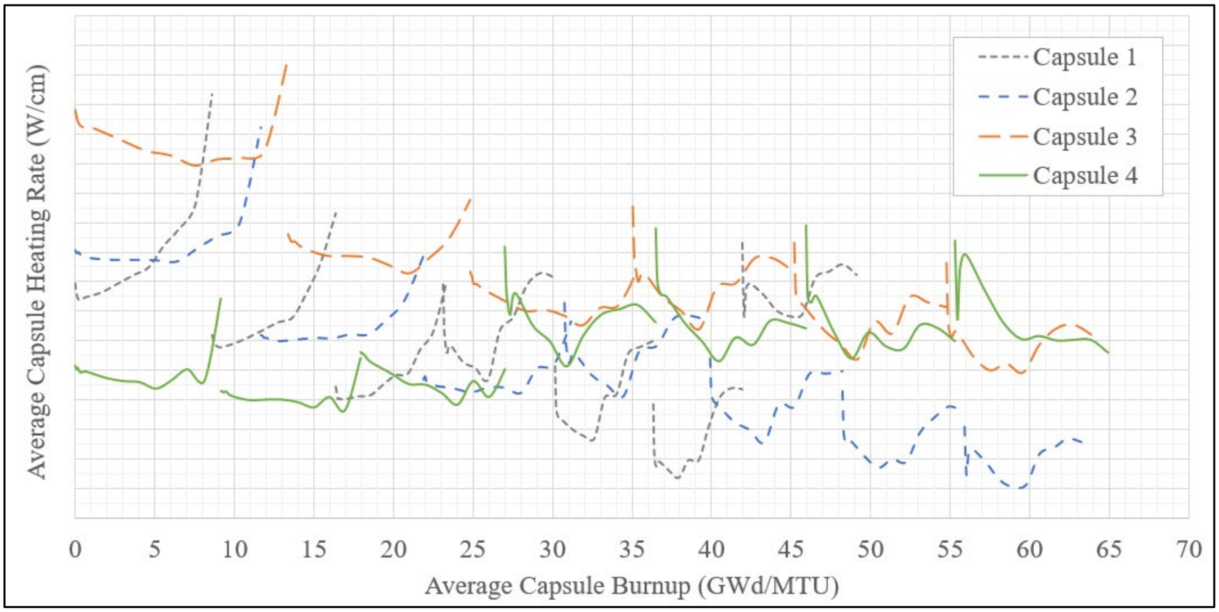

Figure 9. Average capsule linear heat generation rate as a function of burnup.

As mentioned in Section 2, the movement of the neck shims in ATR have an effect on the experiment heating rates, which can be seen in Figure 9. While the capsules are in Inner-A positions (irradiation cycles 1–3), there is a spike in the heating rate near the end of each cycle when the neck shims nearest the experiment are withdrawn. Conversely, when the capsules are in Outer-A positions (irradiation cycles 4–7), the rise in heating rate comes much earlier in the irradiation cycle because the neck shims nearest the Outer-A positions are withdrawn first. This is a unique effect for drop-in irradiation tests in ATR and not necessarily indicative of heating rate profiles in PHWR/CANDU environments.

3.2. Thermal Analysis

The heat transfer design of the CCTE-ANEEL-1A capsule is dependent on two gas gaps to control the fuel temperatures. The most influential gap is between the rodlet cladding and outer capsule, which experiences the highest temperature gradient. The least influential gap is the distance between the experiment pellet outer diameter and the rodlet inner diameter. Both gaps are filled with helium, which is a good conductor of heat between surfaces. The gap distance between the rodlet cladding and outer capsule is optimized to achieve the target peak internal fuel temperature for each capsule based on the heating rates provided in Section 3.1.3. Given the spread in the beginning of life heating rates, temperature targets were established near the middle of the irradiation (30 GWd/MTU), where the heating rates are much more stable. Table II shows the peak temperatures for fuel, rodlet cladding material (SS316H), and outer capsule material (SS316L) for each of the capsule types at 30 GWd/MTU.

Table II. Peak predicted capsule temperatures at 30 GWd/MTU.

Capsule Number	Peak Internal Fuel Temperature (°C)	Peak Rodlet Temperature (°C)	Peak Capsule Temperature (°C)
1	1352	383	137
2	1167	530	125
3	1201	545	131
4	1173	532	127

To simplify the gap conduction equation, all outer capsules are pressurized to 100 psia, which will essentially remove the need to account for the jump gap distance. The additional pressure increases the number of gas molecules and mean free path distance. Additional details on gap conduction and jump gap are explained in detail by Toptan [1]. The heat transfer models used for the thermal analysis account for both the degradation of thermal conductivity due to irradiation as well as the amount of fission gas released into the rodlet gas plenum. An example of the 2D-axisymmetric heat transfer model is provided in Figure 10.

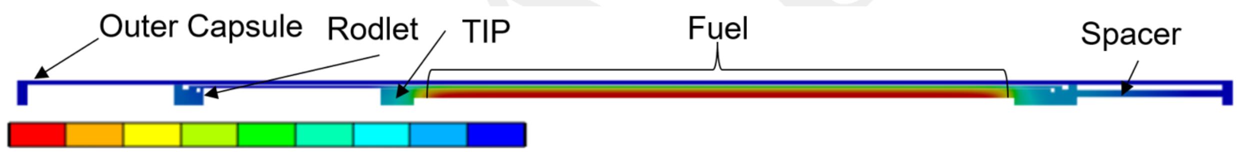

Figure 10. Temperature contour plot of a generic CCTE-ANEEL-1A capsule.

3.3. Structural Analysis

The experiment outer capsules are designed to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) as a Section III, Class 1 Pressure Vessel [2]. This ensures that the experiment remains isolated from the ATR coolant in the event of a breached rodlet, which is a safety requirement for irradiation in ATR. The outer capsules are evaluated based on stresses resulting from external pressure from the reactor coolant, internal pressure from experiment fission gas release assuming a breached rodlet, and thermal expansion using temperature profiles from the thermal analysis.

The rodlets, while not required to be an ASME Section III pressure boundary, are evaluated using ASME BPVC Section III, Division 5 [3] as a guide. This division of the code specifically considers high-temperature pressure vessels. Since the rodlets are expected to be above 800°F (427°C) they are potentially susceptible to thermal creep deformations. The rodlets are evaluated for both inward creep due to external pressure from initial pressurization of the capsule plenum and outward creep due to internal pressure resulting from fission gas release. The evaluation is done in 10-day increments for seven 60-day cycles using conservative assumptions for both temperature and pressure. At each increment evaluated, the rodlets are within design and service level load-controlled stress limits. In addition, the rodlets meet simplified criteria for strain limits meant to limit creep strain in the rodlets. Therefore, inward creep of the rodlets at beginning of life and outward creep of the rodlets at end of life are expected to be negligible.

4. CONCLUSIONS

Design of an ATR irradiation experiment to test the limits of the CCTE-designed, ANEEL fuel has been completed. The CCTE-ANEEL-1A experiment will feature three test trains, each with four capsules containing pellets with varying ratios of $(Th,U)O_2$. Burnup targets for each test train were chosen to assess the potential benefits of ANEEL fuel across a range of burnup intervals. Irradiation is expected to begin in

Spring 2024 and reach the burnup targets of 20, 40, and 60 GWd/MTU. Following irradiation, the capsules will cool in the ATR canal and then be shipped to the INL HFEF for PIE.

ACKNOWLEDGMENTS

Physics calculations in this report are made using the MC21 Monte Carlo code [4], developed and maintained by the Naval Nuclear Laboratory. Analyses in this report made use of the resources of the High-Performance Computing Center at Idaho National Laboratory, which is supported by the Office of Nuclear Energy of the U.S. Department of Energy and the Nuclear Science User Facilities under Contract No. DE-AC07-05ID14517.

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