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FUEL PERFORMANCE STUDIES AT IDAHO NATIONAL LABORATORY MAKING USE OF THE BYRON FUEL SHIPMENT

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Changing the World's Energy Future

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ABSTRACT

In December of 2023 a shipment of commercially irradiated fuel rods from the Byron Generating Station in Illinois was successfully shipped to the Materials and Fuels Complex (MFC) at Idaho National Laboratory (INL). The make-up of the rods includes a mix of cladding types from traditional Zirconium alloy cladding to advanced Zirconium alloy, and chrome coated Zirconium Alloy. Burnups range from regular end of life values to over 70 GWd/MTU rod average. The R&D plan for the rods involves multiple projects from developing licensing data for new claddings to integral transient tests to support burnup extension efforts in the United States. The R&D began in early 2024 with the non-destructive examinations of the rods after which they will be sectioned for microscopy, mechanical testing, and analytical chemistry. Additionally, many rod segments will be refabricated into new test pins and inserted into a static water capsule for integral Reactivity Initiated Accident (RIA) and Loss of Coolant Accident (LOCA) testing at the TREAT reactor.

1 Introduction

Following the 2011 Great Tōhoku earthquake and tsunami that caused extensive damage to the Fukushima Daiichi nuclear power station in Japan, the U.S. Department of Energy (DOE) launched the Accident Tolerant Fuel (ATF) program with the goal of developing fuels for existing light water reactors (LWRs) that could better withstand a loss of cooling event [1]. To ensure economic competitiveness of the new fuel forms, the U.S. nuclear industry has stated their goal to develop ATF concepts with extended enrichment (>5%) and burnup limits (up to 75 GWd/MTU rod average) [2]. Thus, understanding the performance of LWR fuels with high rod average burnups, particularly in design basis accidents, has also become a critical part of the U.S. ATF program.

A key enabling objective for the U.S. national laboratories has been the revitalization of LWR fuel testing infrastructure, particularly at Idaho National Laboratory (INL) and Oak Ridge National Laboratory (ORNL). Key capabilities start with the ability to receive, unload, nondestructively and destructively examine full length ATF rods that were irradiated as part of lead test rod (LTR) or lead test assembly (LTA) irradiation campaigns in commercial reactors. Further capabilities include the refabrication of test pins from the commercial rods and their re-irradiation in transient reactors, and steady state test reactors. An initial shipment of LTR material was made from the Byron Generating station to ORNL in the spring of 2021 and contained some high burnup material and first cycle ATF materials. Results from this campaign are also being reported at this Topfuel conference [3]. The first shipment to INL of commercially irradiated ATF LTRs was recently received and unloaded at INL's Materials and Fuels Complex (MFC) in January of 2024 (Figure 1). The shipment was made using the Nuclear Assurance Corporation International's (NAC) legal weight truck (LWT) cask and was a multi-program shipment that contained 25 commercially irradiated fuel rods with both ATF and standard designs.

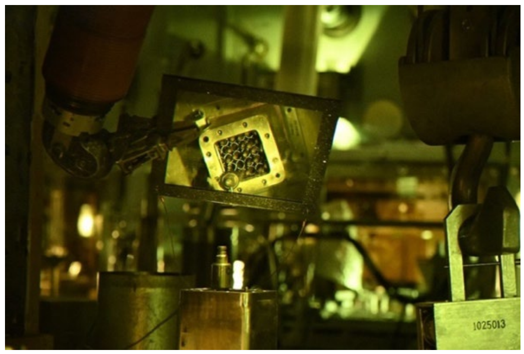


Figure 1. Unloading of a fuel rod from NAC-LWT basket

2 Characteristics of the Byron Fuel Rods

The 25 rods in the shipment from Byron to INL consist of 3 families of cladding varieties over a range of burnups with most of the rods being near end of life or above existing burnup limits. The cladding types consist of traditional Zirconium alloy cladding type rods, Westinghouse's leading ATF fuel rod design which is a cold spray chrome coated Zirconium Alloy rod [4], and fuel rods that consist of an advanced zirconium alloy cladding. Figure 2 shows the makeup and distribution of the rods according to their cladding type and calculated rod average burnup level. Calculated burnups will be confirmed in post irradiation examination.

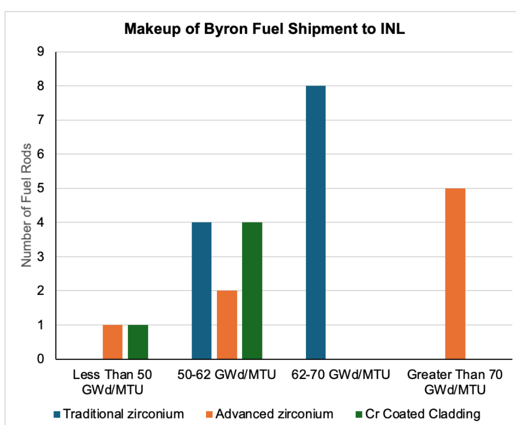


Figure 2. Makeup of fuel rods in Byron shipment

Of the 25 rods in the shipment, only 13 will be used for fuel performance studies. The other 12 are part of an electro-chemical reprocessing demonstration project between the U.S. and the ROK [5]. Four of the thirteen rods being used for fuel performance studies are part of a bi-lateral commercial project between Westinghouse and INL to investigate the performance of the advanced zirconium claddings included in the shipment. One of these rods is being shared between the commercial project and the ATF project resulting in 10 total rods that will be utilized within the ATF R&D program. The make-up of the 10 rods used by the ATF R&D program are shown in Table 1 below. The center ~3m of the rod has a relatively flat burnup profile (plateau region) with only small depressions near the assembly grid spacers and intermediate flow mixer (IFM) locations. The plateau region burnup is usually between 7% and 10% higher than the rod average. Figure 3 below shows the calculated axial burnup profiles of the 10 rods being investigated as part of the ATF project.

Table 1. Details of fuel rods to be used for fuel performance testing.

Rod ID	***Burnup Rod Avg (GWd/MTU)***	***Cladding***	***Fuel***	***No. Of Cycles***	***Discharge Date***
9EU	62.7	Zirconium Alloy	UO_2	3	Spring 2005
HI5	72.4	Zirconium Alloy	UO_2	3	Fall 2011
KBV	68.6	Zirconium Alloy	UO_2	3	Spring 2003
4NB	65.8	Zirconium Alloy	UO_2	3	Spring 2003
SXM	71.4	Zirconium Alloy	UO_2	3	Fall 2011
Q8V	54.8	Cr Coated Zirconium Alloy	UO_2	2	Spring 2022
4DS	54.6	Cr Coated Zirconium Alloy	Doped UO_2	2	Spring 2022
RK3	29.1	Cr Coated Zirconium Alloy	UO_2	1	Fall 2020
FM5	55.1	Cr Coated Zirconium Alloy	UO_2	2	Spring 2022
1D6	55.2	Zirconium Alloy	UO_2	2	Spring 2022

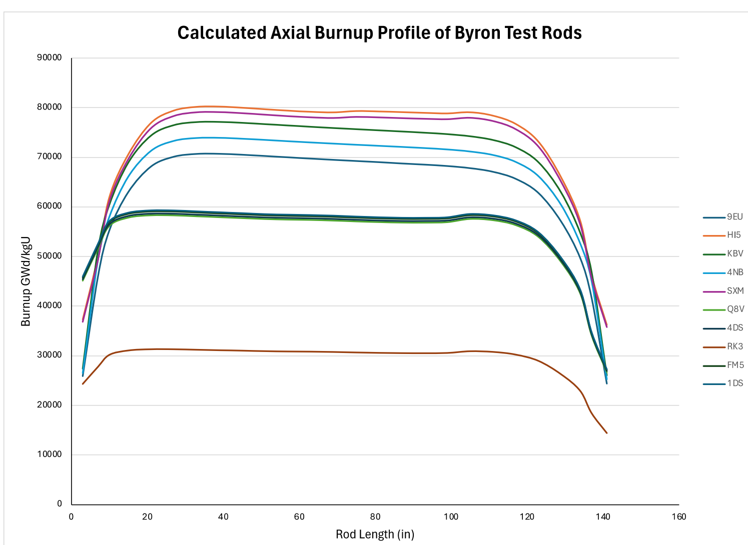


Figure 3. Calculated axial burnup profile of fuel rods used for fuel performance studies.

3 Post Irradiation Examinations

The post irradiation examinations begin with nondestructive examinations which consist of detailed visual examinations, an axial gamma scan, and a profilometry measurement of the rod diameter at several angles. The facility where the rods are received and where most of the examinations occur is called the Hot Fuels Examination Facility (HFEF). In HFEF the rods are handled in the vertical orientation. While this is advantageous as it is the orientation that the rods are irradiated in, the examination equipment in is only tall enough to handle a little over 3.2m of the rod. To examine the very top of the rod, the rods need to be inverted. An inversion fixture was designed and developed for the hot cell to flip the commercial rods 180 degrees so they can be handled upside down and the full length can be examined. The inversion fixture is also necessary to support fission gas puncture and rough sectioning at the beginning of the destructive examinations. For the commercial rods in the Byron shipment, the most important information from the nondestructive examinations comes from the part of the rod where the burnup profile is relatively flat, the plateau region. All of the plateau region can be examined without inverting the rod, thus no rods have been inverted thus far in the nondestructive examinations and there are no current plans to do so. Figure 4 below shows the results of the gamma scan and profilometry measurements from rod 9EU. The source of the diameter change in between grids 3 and 4 is currently undetermined and will be examined in destructive examinations via metallography.

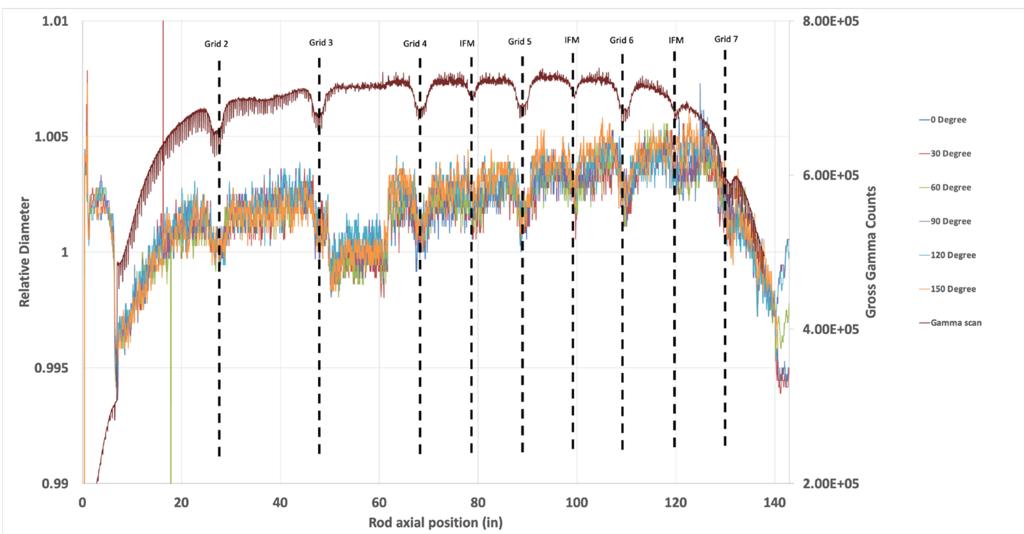


Figure 4. Gamma scan and profilometry measurements of fuel rod 9EU

Destructive examinations begin with a rod puncture followed by rough sectioning. The rod puncture test also conducts a series of expansions and evacuations to determine both the rod internal pressure and free volume, while samples of the rod gas is collected for quantitative gas analysis by gas mass spectrometry. The rough sectioning is required to provide samples that are less than 1.3 meters in total length so they can be laid out horizontally in a dedicated sectioning and containment box which is a separate enclosure inside the hotcell (Figure 5). Rough sectioning occurs with the rod inside the inversion fixture with the rod laid horizontally going back into the interior of the hot cell. Cuts are made with a modified pipe cutter, the accuracy of which is around 5mm and renders a small section on either side of the cut unsuitable for further testing. Once the rough sections are in the containment box, more precise measurements with accuracy of <1mm can be made. The rod's rough sections can be further sectioned using a slow speed diamond bladed saw. The fine sections are prepared for 1 of 4 purposes, (1) metallography/ceramography mounts, (2) analytical samples for cladding light element analysis (oxygen/hydrogen) and fuel dissolution and radiochemical assay by inductively coupled plasma mass spectrometry (ICP-MS) of the fuel to determine burnup and heavy metal inventory, (3) samples for mechanical or thermophysical property examination, or (4) refabricated test pins for subsequent transient or steady state irradiation in one of INL's test reactors. Some segments will also be shipped to ORNL as part of a material exchange to support additional out-of-pile testing using the Severe Accident Testing Station (SATS). A sample cutting plan centered around preparing a test pin for transient irradiation is shown in Figure 6.

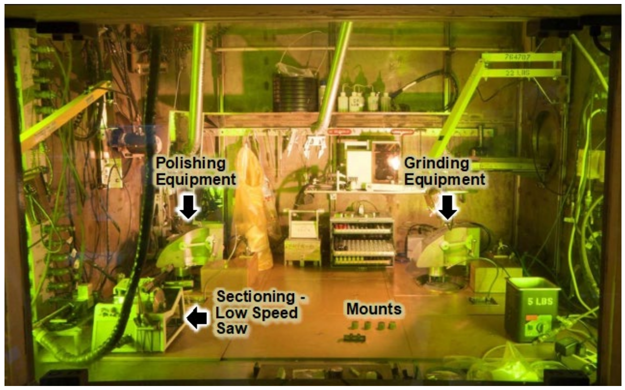


Figure 5. The containment box at HFEF

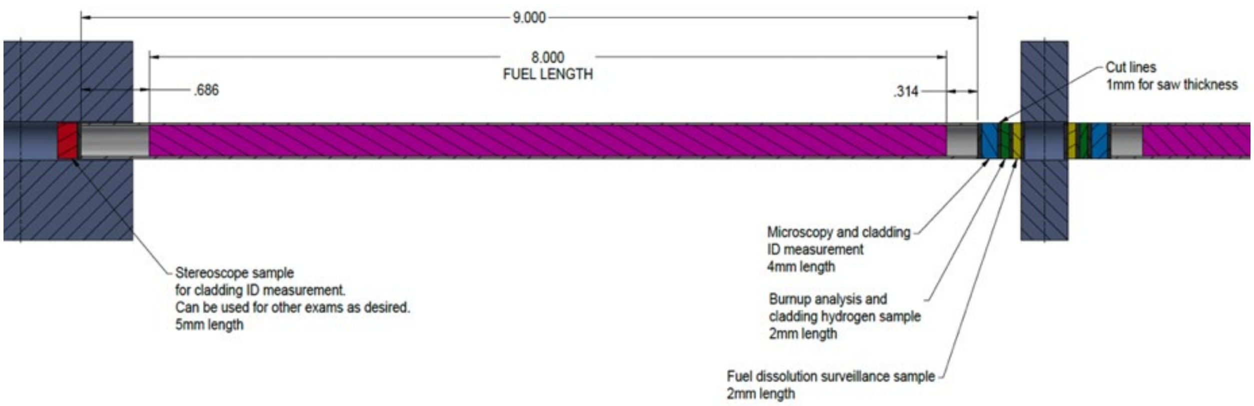


Figure 6. Draft sectioning plan to extract a segment for future in-pile experiment.

Metallography samples can be examined using optical microscopy in another nearby enclosure in HFEF called the metallography box. Micro hardness testing can also be performed in the metallography box. However, electron microscopy takes place in another dedicated facility at MFC called the Irradiated Materials Examination Facility (IMCL). Likewise analytical samples are sent to a dedicated analytical laboratory at MFC.

Mechanical testing to determine the elasticity, strength, and ductility of the new ATF claddings after irradiation is an essential step toward qualification of these cladding and fuel systems. INL have previously developed ring and axial tensile testing techniques for cladding tubes from test pins irradiated in the advanced test reactor [6][7]. These techniques are equally applicable for cladding samples harvested from commercially irradiated test rods. The additional volume of material available in commercial rod affords other options for mechanical testing as well. INL has developed an internal pressure test to determine anisotropic mechanical models for tubular cladding materials using multi-axial loading [8][9]. The system is being modified and adapted currently to function in the HFEF hotcell. A key part of the systems adaptation will be moving from a pneumatic pressure test to a hydraulic pressure test using a high temperature liquid metal to enable testing up to 450 °C. Figure 7 below shows a schematic of hydraulic burst test being developed.

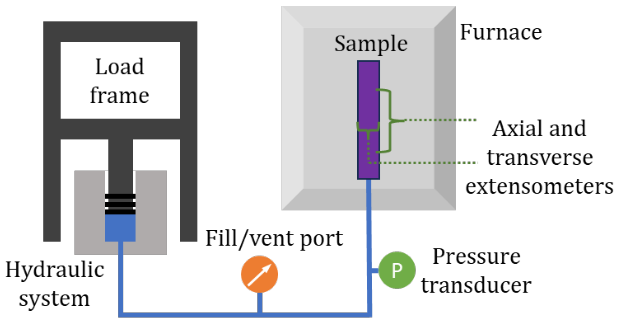


Figure 7. Burst test schematics for both 'Benchtop' and 'Hotcell' systems

4 Transient Irradiations Supporting International Fuel Safety Research Programs

Understanding the performance of high burnup fuel, both standard and ATF designs, in design basis accidents is a key part of the DOE's ATF program. The U.S. Nuclear Regulatory Commission (NRC) has stated that more information is needed on the performance of high burnup fuel in both Reactivity-Initiated accidents (RIAs) [10] and Loss of Coolant Accidents (LOCAs) [11]. A key part of addressing these concerns has been the restart of the TREAT reactor at INL [12] and the resumption of water capsule based transient testing of LWR fuel rods [13]. Figure 8 below shows a comparison of results between the recent water capsule commissioning campaign at INL and historical tests that were part of the SPERT-IV program.

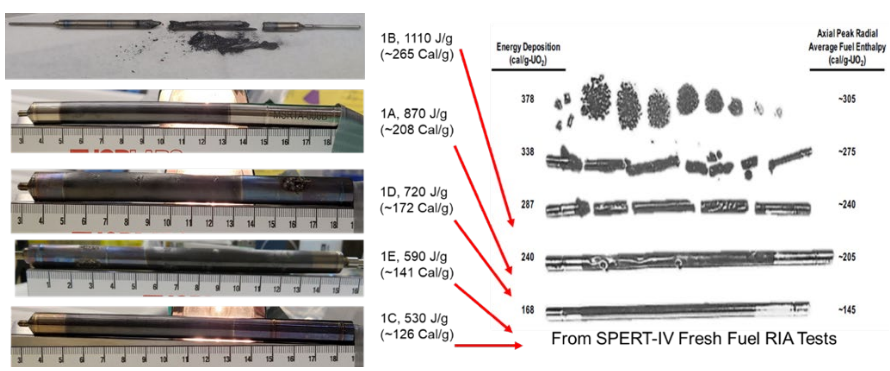


Figure 8. Outcome of TREAT water capsule tests compared to SPERT-IV tests of similar conditions.

For reactivity-initiated accidents existing programs at the NSRR reactor in Japan [14] and the CABRI reactor in France [15] have shown a vulnerability of high burnup fuels to fail early due to pellet cladding mechanical interaction (PCMI). This is primarily due to the precipitation of brittle hydride formations in the zirconium alloy claddings as the result of waterside corrosion. New fuel rod designs, particularly ATF designs, will have significantly less corrosion and thus hydrogen uptake at high burnup. The performance of LWR fuel rods with high burnup oxide fuel pellets but with low hydrogen and low oxide claddings has not been extensively researched and it is likely that more testing is needed to determine the validity of existing fuel rod failure limits [16]. To address this testing need INL, in partnership with Westinghouse, NRC, the Japan Atomic Energy Agency (JAEA), and the French Institute for Radiation Protection and Nuclear Safety (IRSN) have launched the High burnup Experiments in Reactivity-initiated Accidents (HERA) project under the Nuclear Energy Agency's (NEA) Framework for Irradiation Experiments (FIDES) program. The HERA project aims to better understand the effect of transient pulse width, fuel burnup, and cladding hydrogen content on PCMI failure limits in RIA transients [17]. The top sections from the plateau regions of rods 9EU, HI5, Q8V, and 4DS will make up the HERA test matrix. The test matrix includes an end of life fuel rod (9EU), a high burnup rod (HI5), an end of life chrome coated cladding rod (Q8V), and a chrome coated cladding rod with doped fuel at end of life (4DS). These rods represent a mix of traditional and advanced cladding types at end of life and extended burnups. The top parts of the rods are selected for the RIA tests as they are where the rod corrosion levels, and thus hydrogen contents, are likely to be the highest.

For LOCA-type transients, research from the Halden reactor [18], as well as in several different hot cell experiments [19] have shown the propensity for high burnup fuel rods which undergo ballooning and bursting in LOCA tests to experience a significant volume of fine fragmentation, relocation, and dispersal (FFRD) of fuel pellet material from the fuel rod. Addressing FFRD in LOCA transients is a key issue facing the nuclear industry and the NRC prior to licensing of LWR fuel products with extended burnups. To help address this research need, INL, and ORNL through a collaborative partnership with the Electric Power Research Institute (EPRI) developed a test plan to conduct a series of 9 LOCA tests in static water capsules at INL's TREAT reactor and a parallel series of tests in hot cell furnaces at ORNL [20]. The tests aim to evaluate the effect of blowdown, fuel temperature evolution, and temperature ramp rate on the propensity of a high burnup fuel rod to experience FFRD in a LOCA transient. To conduct LOCA tests in a transient reactor water capsule a blowdown tank was added on to a traditional water capsule. Timing of the transient reactor power with the activation of the blowdown valve allows for a simulation of a LOCA transient starting from a representative full power condition, through the blowdown and heat up phase of the transient. The first 4 of the planned 9 LOCA transients have also been accepted by the international LWR fuels research community into the NEA FIDES program within the LOC-HBu project. These 4 tests will use material from the 9EU rod and HI5 rod. Two rod segments from 9EU at end-of-life rod average burnup, will be tested. The first test will be in a traditional, decay energy heat up manner with a slower temperature ramp. The second will be a more representative, combined stored energy + decay energy heat up test with an initial faster cladding temperature ramp while the vessel is still blowing down, followed by the slower temperature ramp (Figure 9). The tests will then be repeated with two rod segments from HI5 which has a higher, extended burnup. Later tests on doped fuel near the FFRD burnup threshold (mid 60 GWd/MTU) will also be explored.

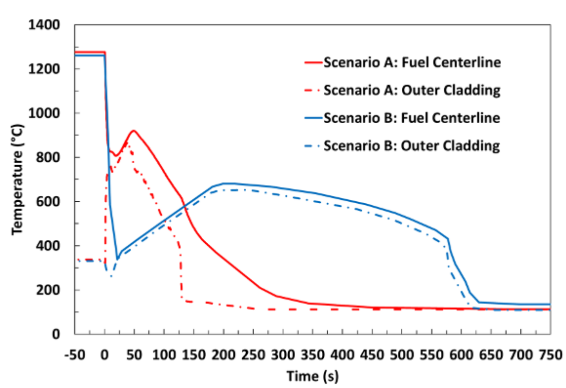


Figure 9. Decay energy heat up (blue) vs stored energy heat up (red)

5 Dynamic Reirradiations to Optimize Fuel Operational Limits

While rods in the Byron fuel shipment do contain rods that have already been irradiated past existing burnup limits, the variety of cladding types in the burnup bins is limited, and none of the rods have been irradiated up to the industry goal of 75 GWd/MTU rod average burnup. Extended irradiations in prototypic pressurized water reactor (PWR) conditions can take place in the center flux trap at the Loop 2A irradiation testing facility of the Advanced Test Reactor (ATR). Irradiations of various ATF concepts have been taking place in this test position since 2018 and have been happening in parallel the LTR/LTA irradiations taking place in commercial reactors. While the burnup accumulation in ATR is slower than in commercial reactors, (primarily due to lower capacity factors) the test reactor irradiations can be performed at higher powers and to much higher burnups. Additionally, the irradiations in the test reactor can be either instrumented or examined poolside more frequently. For the initial high burnup re-irradiations in ATR, the plan is to incorporate an integral plenum pressure bellows into the rod design such that the plenum pressure, along with rod diameter, and rod burnup, can be nondestructively examined periodically during the re-irradiation every 120 – 180 days. The test pins which undergo reirradiation will be ~25cm long with a 16cm fuel column. The test pins will be grouped into a 2x3 array and 4 such arrays will be stacked axially in the ATR core (Figure 10). Test pin powers of up to 400 W/cm will be allowed and achievable even in the high burnup pins.

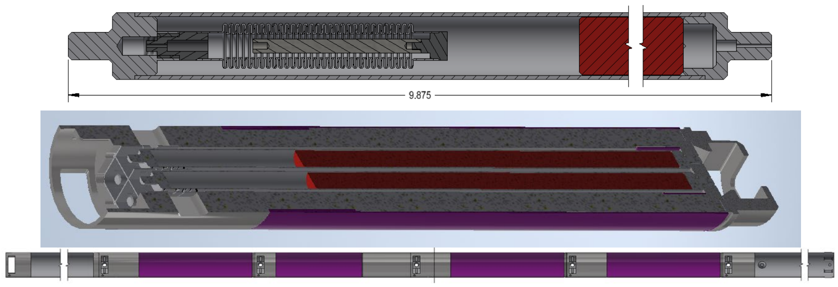


Figure 10. Schematic of ATR test pin (top), test assembly holder (middle), and full test train (bottom).

In addition to the steady state tests, INL is preparing to conduct power ramp testing in the center flux trap of ATR using a power axial locating mechanism. The purpose of the tests is to evaluate failure thresholds and fuel performance behavior of coated cladding and doped fuel at multiple burnup levels in power ramp conditions. Key behaviors in addition to failure limit are cladding permanent hoop strain following ramp and transient fission gas release. The tests would take place during shorter, higher power cycles that occur at ATR 1-2 times per year. The test design is still in the conceptual phase but would involve the simultaneous ramping of three test pins from a standard conditioning power level to three different ramp terminal power levels. This effect is achieved by taking advantage of the inherent flux tilt in the ATR core during the high-power cycles and through the placement of hafnium shields in the test train to shape the flux around the specimens as shown in Figure 11 and Figure 12. Hoop strain and fission gas release will be measured nondestructively in the canal after the ramp test using the same techniques described for the steady state irradiated rods.

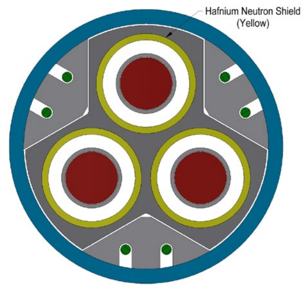
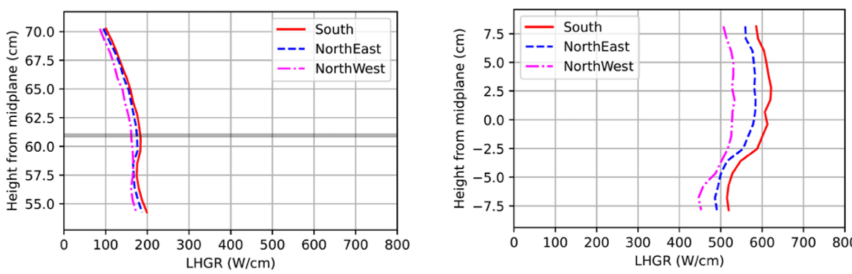


Figure 11. Cross section of ramp test design for the center flux trap at ATR including layered Hf shield configuration.



Top of the Core Middle of the Core

Figure 12. Axial power profiles of the ramp test pins at conditioning and ramp terminal power levels.

6 Conclusions

The shipment of commercially irradiated nuclear fuel rods from the Byron Nuclear Generating station to INL was successfully completed and nondestructive examinations have begun on the rods. Examinations will continue on the rods over the next several years, supporting both bilateral fuel qualification as well as multilateral and multinational fuel safety programs. Test segments will be remanufactured for use within several re-irradiation programs as well as out-of-pile integral tests at ORNL. The first re-irradiation programs will consist of transient irradiations in the TREAT reactor simulating RIA and LOCA conditions with the goal demonstrating the performance of HBU fuels in design basis accidents. The transient reactor test projects are part of the NEA FIDES program. Later reirradiations to even higher burnup and under power ramping conditions will take place in the center flux trap of the ATR reactor.

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