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Status of Wrought FeCrAl-UO_2 Capsules Irradiated in the Advanced Test Reactor

Nuclear Technology Research and Development

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SUMMARY

Candidate cladding materials for accident tolerant fuel applications require extensive testing and validation prior to commercial deployment within the nuclear power industry. One class of cladding materials, FeCrAl alloys, is currently undergoing such effort. Within these activities is a series of irradiation programs within the Advanced Test Reactor. These programs are developed to aid in commercial maturation and understand the fundamental mechanisms controlling the cladding performance during normal operation of a typical light water reactor. Three different irradiation programs are on-going; one designed as a simple proof-of-principle concept, the other to evaluate the susceptibility of FeCrAl to fuel-cladding chemical interaction, and the last to fully simulate the conditions of a pressurized water reactor experimentally. To date, non-destructive post-irradiation examination has been completed on the rodlet deemed FCA-L3 from the simple proof-of-concept irradiation program. Initial results show possible breach of the rodlet under irradiation but further studies are needed to conclusively determine whether breach has occurred and the underlying reasons for such a possible failure. Further work includes characterizing additional rodlets following irradiation.

CONTENTS

FIGURES

ACRONYMS

ATF	Accident tolerant fuel
ATR	Advanced Test Reactor
FCCI	Fuel-cladding chemical interaction
FeCrAl	Iron-chromium-aluminum
FMDP	Fissile Materials Disposition Program
GASR	Gas Assay, Sample, and Recharge
GE	General Electric
HFEF	Hot Fuel Examination Facility
HFIR	High Flux Isotope Reactor
LG	Large grain
LHGR	Linear heat generation rate
LOCA	Loss of coolant accident
LWR	Light water reactor
MFC	Materials and Fuels Complex
MOX	Mixed oxide
PWR	Pressurized water reactor
SG	Small grain
SiC	Silicon carbide
UO_2	Uranium dioxide
VIM	Vacuum induction melt

STATUS OF WROUGHT FeCrAl-UO_2 CAPSULES IRRADIATED IN THE ADVANCED TEST REACTOR

1. INTRODUCTION

Iron-chromium-aluminum (FeCrAl) alloys are a promising class of ferritic Fe-based alloys for accident tolerant fuel (ATF) cladding applications due to their superior oxidation and corrosion resistance in simulated normal and off-normal conditions within a nuclear power reactor [1–3]. Additionally, the FeCrAl alloy class has been demonstrated to have high resistance to radiation-induced swelling, stress corrosion cracking resistance, and good weldability and fabricability [4–7]. These properties and attributes have been primarily demonstrated on sheet product and simulated environments where integral effects have not been widely studied or understood. The result is a limited operational knowledge of FeCrAl alloys under prototypical environments for light water reactors (LWRs) using thin-walled cladding. Additionally, only the early studies on model GE alloys [8] has determined the influence of UO_2 fuel on the performance of FeCrAl alloys when co-irradiated and showed little interaction at temperatures below 1000°C.

To supplement the current database on FeCrAl alloys, a series of irradiation programs have been conceptualized and currently on-going using the High Flux Isotope Reactor (HFIR) and the Advanced Test Reactor (ATR). The HFIR irradiations are primarily focused on alloy screening experiments and mechanical properties evaluations following high-intensity irradiation. Reports on this program can be found elsewhere [9–13]. The ATR irradiations are focused on progressing towards integral tests and fueled experiments. Three separate irradiation designs are being used within the ATR for evaluation of wrought FeCrAl alloys. Additional studies are focused on powder metallurgy derived FeCrAl alloys but falls out of the scope of the work presented here. The three studies have been deemed ATF-1 loss-of-coolant accident (LOCA), ATF-1 fuel-cladding chemical interaction (FCCI), and ATF-2.

ATF-1 LOCA was the first series of ATR testing and was developed as a robust and efficient test train to deploy drop-in capsule experiments containing FeCrAl cladding which encapsulates prototypical commercial pressurized water reactor (PWR) UO_2 fuel. Of particular interest within ATF-1 LOCA, is the performance of the clad including the structural stability and the weldment performance. The follow-on study to the ATF-1 LOCA study is the ATF-1 FCCI study. This study deviates from the standardized rodlet geometry and uses specialized sub-assemblies to determine the direct interaction of UO_2 fuel with varying candidate FeCrAl claddings. This effect is not observable in the ATF-1 LOCA studies due to limited cladding-fuel gas gap closure within the irradiation conditions. The ATF-1 studies are followed up with the ATF-2 studies. The primary goal of the ATF-2 studies are to demonstrate ATF candidate cladding and fuel viability under expected PWR conditions for neutron flux and corrosive media. Such activities would help demonstrate an individual concept's viability thus enabling further commercial maturation of the concept.

The following report highlights the on-going activities related specifically towards wrought FeCrAl alloys in relation to the ATF-1 and ATF-2 studies.

2. SUMMARY OF CAPSULE DESIGNS

2.1 ATF-1 LOCA Rodlets

The ATF-1 LOCA rodlets were developed to demonstrate the ability to deploy novel accident tolerant fuel-clad systems within the neutron environment and provide valuable test specimens for LOCA-based testing after irradiation. Additionally, the rodlets provide the ability to determine the mechanical and physical behavior of the candidate cladding and weld performance under neutron irradiation in the presence of fuel.

The LOCA rodlets were developed based on a miniaturized PWR fuel rod. This design enables the use of prototypical PWR fuel within the rodlets. The overall design and components are shown in Figure 1. The rodlets include a top cap, bottom cap, cladding tube, UO_2 fuel, retention spring, and SiC thermometry discs. SiC discs are added to verify modeled irradiation temperatures have been met. The assembly design is a drop-in capsule design based on the mixed-oxide (MOX) fuel irradiation design previously used within the DOE Fissile Materials Disposition Program (FMDP) [14]. As such, the entire irradiation assembly includes the rodlets which are encapsulated in a capsule assembly. The result is the miniaturized rodlets are not in contact with the primary coolant of the ATR. The lack of coolant-clad interactions prevents significant clad creep-down under irradiation and any evaluations of corrosion from the primary coolant under irradiation.

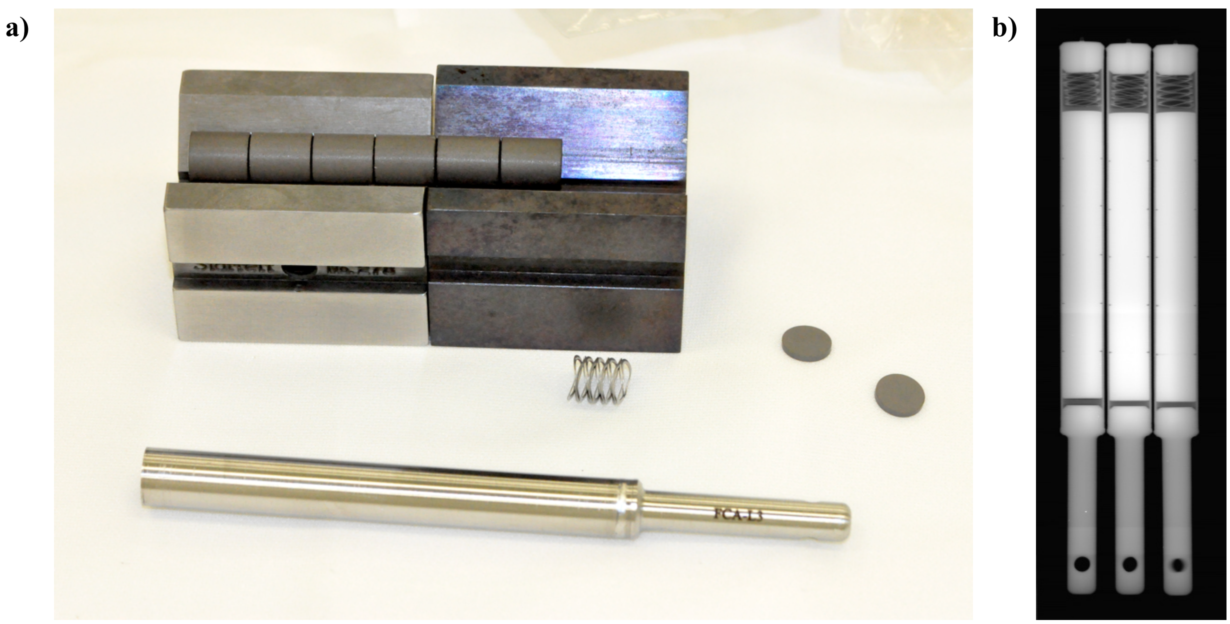

Figure 1: Representative ATF-1 LOCA rodlet components (a) prior to assembly and (b) after assembly using radiography. Rodlet lengths are 4.529 in. nominal.

The assembly configuration was inserted into the small-I positions within ATR. The generalized design allows for a linear heat generation rate of 160-230 W/cm and a nominal cladding temperature of ~350°C. The overall fast flux within the positions is expected to be low (~$3.2x10^{12}$ n/cm^2-s) resulting in a low accumulated damage dose over the lifetime of the experiment. A Generation II FeCrAl alloy, C35MN, was selected for irradiation. Cladding tubes were developed using gun-drilling techniques. At the time of deployment, C35MN was a primary alloy of interest but has been superseded by C26M in later studies within the program. A total of three capsules were built and deployed with end targets of 10, 30, and 50 GWd/MT.

2.2 ATF-1 FCCI Rodlets

The ATF-1 FCCI rodlets were envisioned to directly determine the interaction of prototypical commercial UO_2 fuel with candidate wrought FeCrAl alloys. The driving force for such studies was based on the limited knowledge for FCCI between FeCrAl and UO_2 in a typical clad-fuel fuel-pin configuration at prototypical LWR irradiation conditions. FCCI is known to weaken cladding through clad thinning due to extensive fuel-clad interactions leading to premature failure of a fuel-rod at lower stresses than those expected without the presence of FCCI. To fill this knowledge gap, a series of capsules were designed, built, and irradiated in the ATR test reactor.

The FCCI rodlets were developed based on a simplified diffusion-couple style miniature test rodlet. The capsule is designed in a manner to provide continuous contact between the candidate cladding specimens and sub-sized fuel slices. The design incorporates several layers of clad-fuel specimens enabling the test of different cladding materials and surface treatments within a single irradiation. A retention ring is used around the fuel-clad specimens to limit radial swelling of the UO_2 fuel. This configuration promotes constant contact between the fuel and clad specimens. Within previous reports this configuration is deemed the “H-cup” assembly. A general schematic of the ATF-1 FCCI capsule is shown in Figure 2. Each H-cup assembly is stacked and at the limiting ends of the stack SiC samples, which serves as passive SiC thermometry, and zirconia insulator disks to limit axial temperature gradients are placed. These are also represented in Figure 2. Gas gaps vary along the length of the capsule to mitigate axial temperature profiles during irradiation.

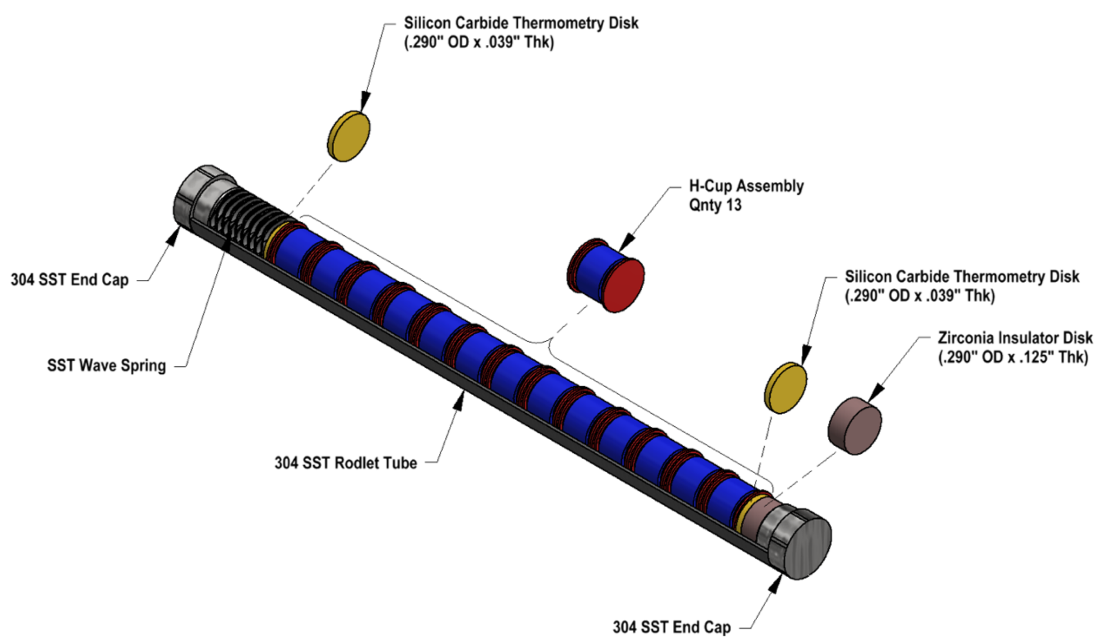

Figure 2: Schematic of ORNL FCCI rodlet. Image courtesy of Nate Oldham (INL) [15]**. Rodlet lengths are 4.75 in. nominal.**

All fuel slices used within this capsule are from the same batch of fuel used within the ATF-1 rodlets to enable cross comparison between the two studies. The fuel discs were manufactured using specialized processes at Idaho National Laboratory (INL) [15]. All discs were enriched to 4.95% and used commercial pressurized water reactor (PWR) blending.

One Generation I, two Generation II, and one commercial alloy were selected as candidate cladding. Details on the delineation between Generation I and Generation II FeCrAl alloys can be found in previous reports of the same program [16,17]. The Generation I alloy, B125Y3 was thermo-mechanically processed using two different techniques to produce a large grain (LG) variant and small grain (SG) variant with identical compositions. This treatment was completed to enable determination of grain size effect on FCCI of FeCrAl, if any. The Generation II alloys include Mo additions, with one variant have a nominal 5 wt.% Al and the other having a nominal 7 wt.% Al; the different alloys provide an opportunity to study composition effects, if any, on FCCI in FeCrAl alloys. The commercial alloy was selected as APMT to provide a reference FeCrAl material for the study. Subsets of one Generation II alloy, C35M3, was pre-oxidized in air at 800°C or 1000°C; all other Generation I and II alloys were pre-oxidized in air at 1000°C. Pre-oxidation was completed to determine the ability of Al_2O_3 as a mitigating barrier to FCCI. The result was thirteen H-cup sub-assemblies in a single capsule, as shown in Figure 3.

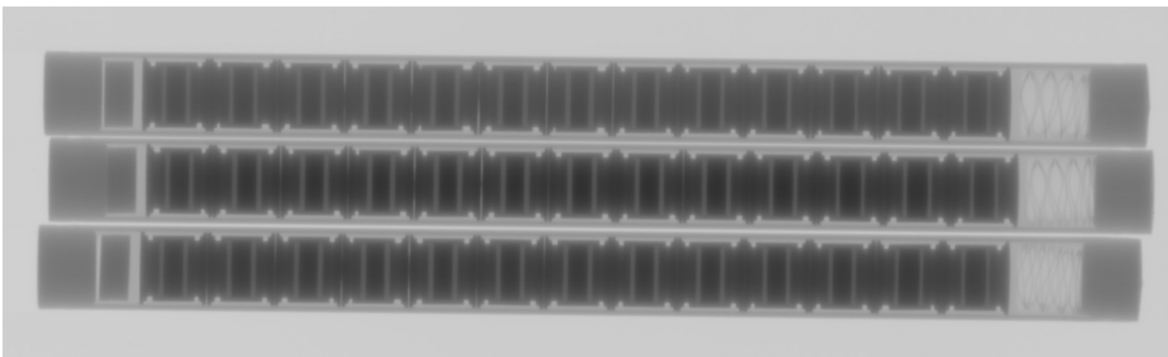

Figure 3: Radiograph of as-built FCCI rodlets prior to insertion in ATR. Image courtesy of Greg Core (INL). Rodlet lengths are 4.75 in. nominal.

Capsules were inserted in the northeast lobe of the ATR in the I-21 irradiation position. Preliminary studies indicate the linear heat generation rate (LHGR) will be ~142 W/cm providing similar power density to those of prototypical UO_2 fuel during LWR operation. Temperatures were designed to mimic those expected at the fuel-clad interface during typical LWR operation. Hence, the predicted design temperatures at the radial center (central node) of the specimens varies in the axial direction between 304°C and 381°C. The temperature at the outer surface (outer node) of the cladding specimens (where the assembly contacts the retention ring) is expected to vary in the axial direction between 295 and 362°C. Three capsules were built and deployed with end targets of 10, 30, and 50 GWd/MT.

2.3 ATF-2 Rodlets

The ATF-2 irradiation program uses a flowing water test loop to enable novel clad and fuel-clad designs to be tested in a neutron environment while simultaneously being exposed to prototypical PWR coolants. Currently, the ATF-2 experimental design is the closest representation of prototypical PWR conditions in the ATR irradiation series. The final design consists of a 2x3 arrangement of one 30.5 cm and four 15.25 cm tiers of positions for rodlets. The tier positioned closest to the top of the reactor is stationary and enables instrumentation of the tier and rodlets within that position. The test train for ATF-2 including the rodlets (in blue) are shown in Figure 4. Not shown is the sophisticated set of sensors that will be deployed to measure irradiation temperature, neutron flux, fuel and clad dimensional changes, and pressure.

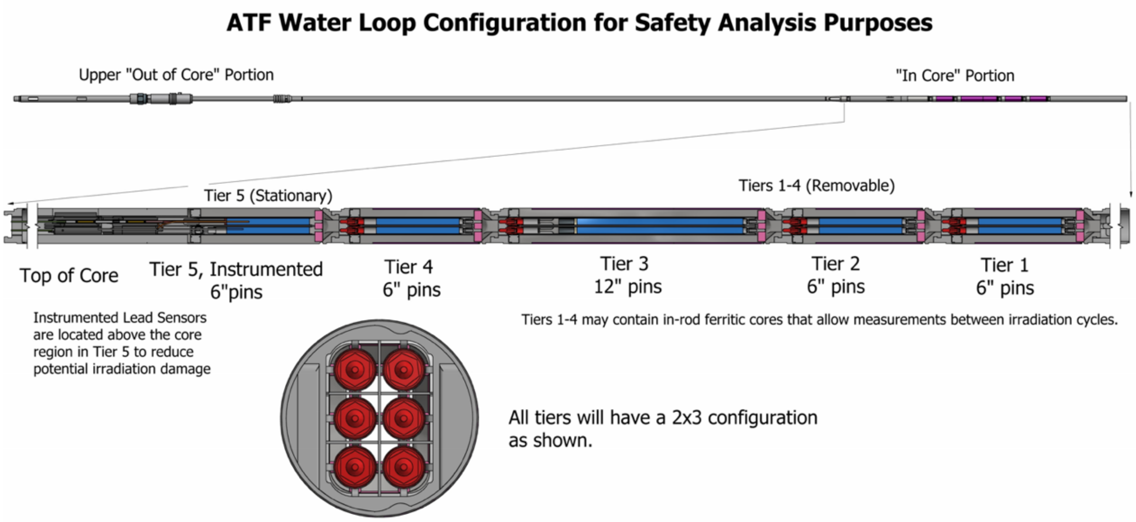

Figure 4: ATF-2 irradiation test train. Figure courtesy of Brian Durtschi (INL) [18].

Two Generation II alloys were provided to enable deployment of the ATF-2 test train within ATR. The first, C06M2, and the second, C36M2, have identical nominal aluminum concentrations but varying in Cr content by 3 wt. %. On-going studies within this program [9,11,13] have indicated Cr content as primary contributing factor to the performance of FeCrAl alloys during normal and off-normal behavior and hence the two variants will enable direct confirmation of these studies. Both alloys were manufactured using traditional wrought alloy practices including vacuum induction melting (VIM) followed by extrusion and annealing at 800°C. Feedstock for production of test specimens using the Generation II alloys have been provided to INL directly.

3. SUMMARY OF CAPSULE IRRADIATIONS

3.1 ATF-1 LOCA Rodlets

Table 1 summarizes the current status of the ATF-1 LOCA rodlets. Rodlets were shipped to INL for insertion into the ATR in September 2014. To date, the lowest target burnup capsule, FCA-L3, has met its target condition and been removed from ATR. This capsule was shipped to the Materials and Fuels (MFC) complex at INL in March of 2017. Post-irradiation examination (PIE) details in regard to this rodlet are provided in Section 4. All other rodlets are currently undergoing irradiation in ATR.

Table 1: Summary and status of ATF-1 FCCI Rodlets

Capsule ID	Rodlet ID	Fuel Type	Cladding Type	Burnup Target (GWD/MTU)	ATR Insertion Cycle	ATR Removal Cycle	Removal Year	Status
ATF-18	FCA-L3	UO_2	C35MN	10	157D-1	160A	2017	PIE
ATF-17	FCA-L2	UO_2	C35MN	30	157C-1	168A	2019	In irradiation
ATF-20	FCA-L5	UO_2	C35MN	50	157C-1	173A	2021	In irradiation

3.2 ATF-1 FCCI Rodlets

Table 2 summarizes the current status of the ATF-1 FCCI rodlets. Rodlets were shipped to INL for insertion into the ATR in September 2016. The later ATR insertion dates of these more complex rodlets compared to their ATF-1 LOCA counterparts means the all rodlets are currently still undergoing irradiation in ATR.

Table 2: Summary and status of ATF-1 FCCI Rodlets

Capsule ID	Rodlet ID	Fuel Type	Cladding Type	Burnup Target (GWD/MTU)	ATR Insertion Cycle	ATR Removal Cycle	Removal Year	Status
ATF-73	FCCI-1	UO_2	Various FeCrAl	10	160B-1	164A	2018	In irradiation
ATF-74	FCCI-2	UO_2	Various FeCrAl	30	160B-1	168A	2019	In irradiation
ATF-75	FCCI-3	UO_2	Various FeCrAl	50	160B-1	173A	2021	In Irradiation

3.3 ATF-2 Rodlets

Feedstock materials to support the Sensor Qualification Test (SQT) and autoclave tests were provided to INL in August 2016. Further development of the ATF-2 test train and testing parameters are currently being completed by INL and General Electric (GE). If needed, ORNL-derived cladding can be provided for use within the ATF-2 test train although not currently planned.

4. SUMMARY OF POST IRRADIATION EXAMINATION ACTIVITIES

As noted in Section 3, the only rodlet currently available for post irradiation examination (PIE) is FCA-L3 which is the 10 GWd/MT ATF-1 LOCA rodlet. The rodlet was shipped from ATR to the Materials and Fuels (MFC) complex at INL. The MFC has been the designated facility to complete preliminary non-destructive evaluation on the capsule and rodlet assembly of the ATF-1 rodlets. Shipment of samples was completed using a GE-100 cask within the second quarter of United States Fiscal Year 2017.

Figure 5a shows the capsule assembly prior to deconsolidation and examination of the rodlet contained within. Optical investigations showed the outer capsule to be in good condition and free of cracks or other noticeable superficial defects. Neutron radiography was performed prior to rodlet extraction to provide initial insight into the condition of the rodlet and fuel pellets, Figure 5b. Radiographs showed no obvious signs of cladding breach within the capsule or rodlet. Axial cracking can be observed within the UO_2 fuel pellets, but such defects are expected within the fuel after irradiation. Additional elements (SiC thermometry, springs, etc.) show no significant change in comparison to pre-irradiation radiographs, Figure 1b.

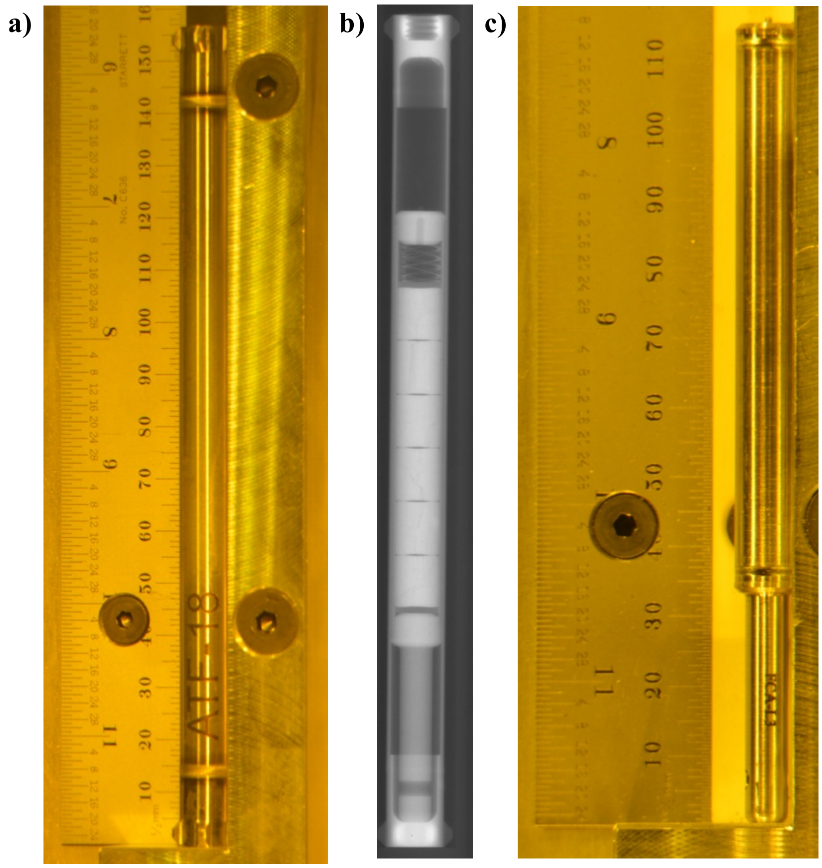

Figure 5: Images of capsule and rodlet assembly for FCA-L3 after irradiation. (a) optical image of capsule assembly, (b) neutron radiography of capsule and rodlet assemblies, (c) optical image of rodlet assembly.

The encapsulating capsule assembly was punctured within the plenum region of the capsule to measure the free volume, the pressure, and gather a plenum gas sample for composition and isotopic analysis. The procedure was completed using the Gas Assay, Sample, and Recharge (GASR) system at the Hot Fuel Examination Facility (HFEF). The measured free volume was determined to be ~3.3 cm^3 while the pressure was deemed identical to the capsule pressure during assembly within the error of the current GASR system. The collected gas sample was determined to contain ^{85}Kr using gamma spectrometry indicating a possible breach and fission gas release from the rodlet into the capsule cavity. The measured free volume is in line with the measured free volume of the rodlet and capsule based on design drawings, further supporting the possibility of breach of the inner rodlet. At the time of this report gas mass spectrometry has not been completed to determine the fission gas content. Given this, based on the irradiation conditions it is anticipated the fission gas release will be below 1%.

The potentially breached rodlet was extracted from the outer capsule to enable further examination of the rodlet. Figure 5c shows a representative high-resolution optical image of the rodlet taken using through window digital photography. Optical images were taken at ~15-30° rotations around the long axis to determine possible cracks or breach locations within the rodlet. Optical images show no visual signs of rodlet breach within the clad or end plugs. Re-inspection of the high-resolution images confirm these observations. Given this, possible flaws still exist and could remain below the resolution limit of the optical inspection equipment. At the time of this report, eddy current measurements are planned to determine any possible flaws/defects not readily revealed using either photography or neutron radiography.

Following non-destructive evaluation, FCA-L3 will be shipped to Oak Ridge National Laboratory (ORNL) for further destructive testing. Due to the radioactivity and isotopics within the fueled rodlet, a specialized cask is needed to aid transportation. Currently, there is no clear path for the shipment of the ATF-1 rodlet from INL to the ORNL IMET facility. Casks with the appropriate DOT approvals to handle the ATF-1 rodlets are either unable to be handled by various hot cell facilities due to the size of the cask and limited infrastructure or casks otherwise have limited to no availability.

Options are currently being explored to address the various programmatic needs for a reliable and readily available cask for transportation of reactor experiment fissile materials between national laboratories. All of the options currently under consideration will require a brand new or amended NRC approved Certificate of Compliance to handle the ATF-1 rodlets and other fuel segment related experiments. A recent analysis by INL Transportation comparing the AOS-050 and AOS-100 casks to the DOE-NE owned Battelle Research Reactor (BRR) cask concluded the BRR cask was the more attractive option for INL considering the overall cost of purchase, procedure development, licensing, and ancillary equipment needs. Based on this input from INL, a similar analysis of the cost of procedure development and ancillary equipment needs associated with the BRR is currently being performed at ORNL for use at the 3525 IFEL hot cell. If procured by INL, the estimated timeline for delivery of the BRR cask and the amended certificate is 12 to 14 months, placing destructive PIE at ORNL of the FCA-L3 rodlet at an estimated timeline of 22-26 months from the time of this report.

5. ACKNOWLEDGEMENTS

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