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Overview of the Water-Based RIA Testing in TREAT

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OVERVIEW OF THE WATER-BASED RIA TESTING IN TREAT

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ABSTRACT

We have completed a series of reactivity-initiated accident commissioning tests with the static water capsule in the Transient Reactor Test Facility. The test campaign included a calibration test followed by five tests in the Static Environment Rodlet Transient Test Apparatus capsule. The conditions varied from room temperature and pressure up to 200°C and 2.5 MPa, with energy depositions varying between ~500–1100 J/gUO_2. The series of tests allowed for a number of instrumentation qualifications and demonstrations, including cladding thermometry, rodlet plenum pressure, cladding elongation, and an electro-impedance boiling detector. This paper documents the design of the capsule and highlights the results from the commissioning tests.

1. Introduction

The mission of the Accident Tolerant Fuel (ATF) Program is to develop the next generation of light-water reactor (LWR) fuel that offers better performance during normal operation and which also exhibits enhanced tolerance to design-basis and beyond design-basis accidents (DBAs and BDBAs, respectively). The goal of the Fuel Safety Research Program within the ATF Program is the identification and quantification of appropriate nuclear fuel safety criteria and thresholds for the ATF materials that are being developed [1].

Reactivity-initiated accidents (RIAs) are a postulated DBA in LWRs. RIAs can happen due to the rapid ejection of the control rods from the nuclear reactor's core, which causes a prompt increase of the fission rate density until the negative temperature feedback shuts down the reactor. The increased fission rate, even for a short period of time, causes a rapid temperature-increase of the fuel. The rapid thermal expansion of the fuel closes the gap between the fuel and the cladding (or both could be already in contact for previously irradiated fuel) and imposes a complex mechanical strain to the cladding. This pellet-cladding mechanical interaction (PCMI) can cause a failure of the cladding early in the transient, but if enough heat is transported from the fuel to the cladding before failure, the cladding temperature increases, and its deformation path changes from strain-driven to internal pressurization at high temperatures, which can cause ballooning and rupture. These failure modes and limits need to be determined for ATF concepts.

Processes occurring during RIAs are complex, and the entire process can best be simulated by integral-test capabilities and prototypic nuclear heating. The Transient Reactor Test (TREAT) Facility of the Idaho National Laboratory (INL) offers the unique capability of conducting integral transient testing. The flexibility of TREAT enables it to support RIA, LOCA, metallic fuel, and separate effect experiments. In preparation for future ATF testing at TREAT, a commissioning test campaign was started to demonstrate the ability for RIA transient testing in a water-based capsule. To support this testing, an experiment capsule termed the Minimal Activation Retrievable Capsule Holder–Static Environment Rodlet Transient Test Apparatus (MARCH-SERTTA) was designed. The purpose of the MARCH-SERTTA commissioning tests was to commission a static water RIA testing capability to demonstrate the ability to deposit a specified amount of energy in a representative test specimen and to observe the key known physics of an RIA transient. This paper will report on the status of the commissioning tests and will not discuss how the results compare to existing data or RIA safety criteria.

2. Experiment Description

2.1 Capsule Design

The commissioning tests were conducted in the MARCH-SERTTA capsules. These capsules are comprised of a capsule top and a capsule bottom and are designed to represent prototypic thermal-hydraulic boundary conditions for the rodlet that are found in an LWR, albeit with static rather than flowing water. A rendering of the capsule is in Fig 1, which points out important design features and instrumentation. The capsule consists of a lower body that houses the fuelled rodlet and instrumentation. The capsule body was filled with water to fully submerge the rodlet. The upper portion of the capsule includes the lid with an expansion volume for rapid water vaporization. The capsule was instrumented with multiple cladding and water thermocouples, an upper bellows system to measure rodlet internal pressure, a linear variable differential transformer (LVDT) to measure fuel rodlet elongation, and an electro-impedance boiling detector. Each rodlet included eight UO_2 pellets, two ceramic insulator pellets (one each at the top and bottom of the fuel pellet stack), clad in a zirconium alloy having dimensions typical of pressurized-water reactors (PWR). The rodlet will be submerged in water at pretest conditions up to 200°C and pressures prototypic of PWR subcooling. A more detailed overview of the test capsule can be found in [2].

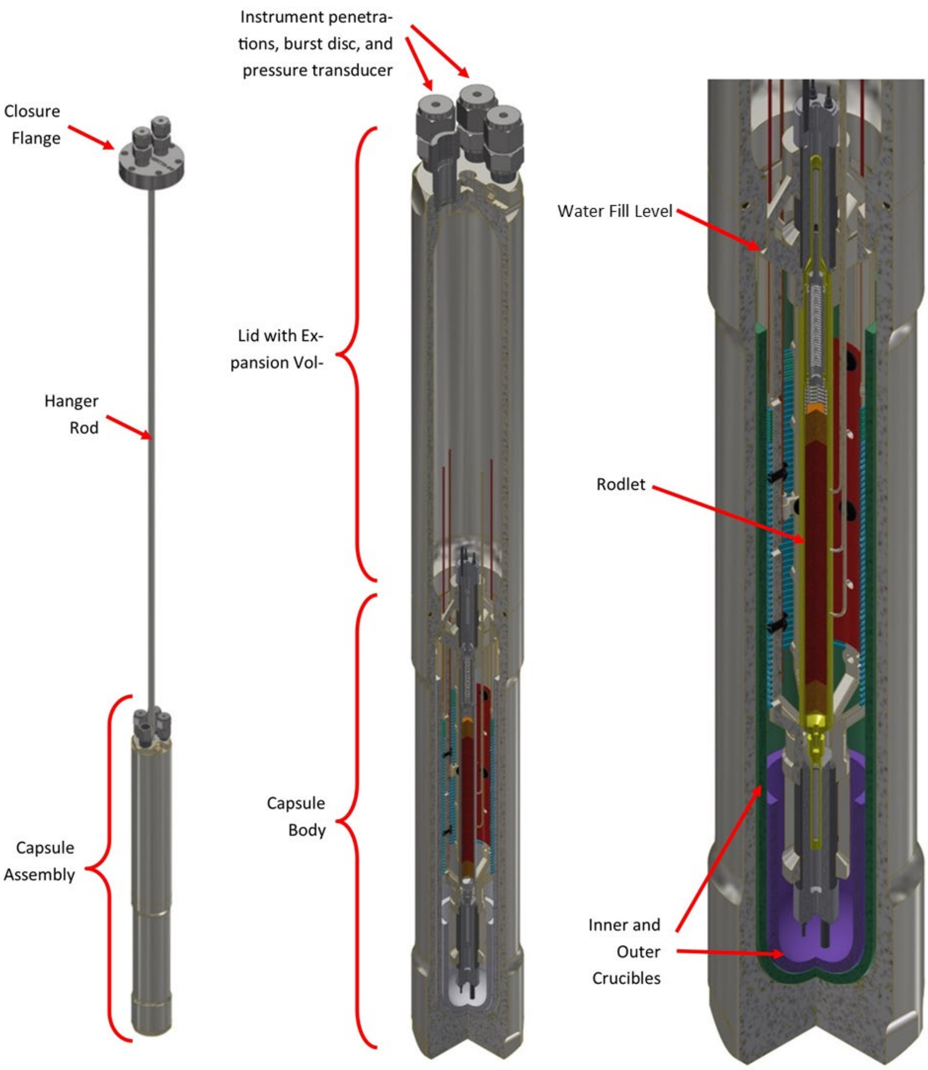

Fig 1. Overview of the MARCH-SERTTA module

2.2 Test Matrix and Objectives

The commissioning campaign included a total of six capsules. The first capsule was the calibration test used to verify the energy coupling between the TREAT reactor and fuel specimen. This is achieved by subjecting the specimen to a low powered transient with a well-defined energy generation in the TREAT core. The fuel rodlet is then sent off for gamma spectroscopy to estimate the number of fissions that occurred during the transient. This can then be used to calculate the TREAT core to fuel energy coupling factor with an uncertainty of 15%. This value can then be used for all subsequent tests to define the heat generation rates in the specimen.

The fuel rod for the calibration transient was constructed with five 0.74% U^{235} pellets and five 4.95% U^{235} pellets. This provided an upper and lower bound of the fuel coupling factor for all subsequent tests in the MARCH-SERTTA commissioning test campaign as well as another test campaign using the same MARCH-SERTTA capsule design.

The remaining five capsules were each subjected to varying energy depositions and starting conditions to target specific phenomena and test specific instruments. The overview of those tests and conditions are summarized in Tab 1. The rodlet pressure is the as-manufactured pressure at ~20°C, and the capsule pressure is the measured pressure at the temperature indicated in the table. Each fuel rod for these remaining tests were constructed with eight 0.74% U^{235} pellets and two ceramic insulator pellets. The respective transient histories and energy deposition plots for the TREAT reactor are shown in Fig 2 for each test.

Test ID	Rodlet Pressure (MPa)	Capsule Pressure (MPa)	Capsule Temperature (°C)	Reactor Energy (MJ)	Specimen Energy Deposition (J/g)	Pulse Width Full-Width-Half-Max (FWHM) (ms)	Targeted Cladding Temperature (°C)	Test Purpose
1-A	0.1	0.1	22	1,272	870	90.5	<1,200	Achieve film boiling from RTP initial conditions
1-B	0.1	0.7	22	1,617	1,110	99.4	>1,200	Observe cladding embrittlement without burst
1-C	0.1	2.23	205	1,042	530	89.8	~850	Achieve film boiling with slightly subcooled initial conditions
1-D	2.0	2.41	207	1,431	720	93.8	<1,200	Demonstrate ballooning and bursting during film boiling
1-E	2.0	1.99	202	1,160	590	89.8	~850	Achieve film boiling with slightly subcooled initial conditions, instrumentation test

Tab 1: ATF-RIA test campaign test matrix in MARCH-SERTTA capsule

2.3 Experiment Models

Throughout the experiment campaign, a variety of modelling tools have been used to provide guidance to the tests and expected outcomes of the experiment. RELAP5-3D [3] was used for thermal-hydraulic predictions and quick parameter studies on the effect of starting conditions on fuel rod temperatures and on whether departure from nucleate boiling (DNB) would be achieved during the experiment. We performed a more detailed thermal-mechanical modelling of the fuel rod to determine possible failure or if clad ballooning may occur with the fuel performance code BISON [4-6]. In some cases, a coupling between BISON and RELAP5-3D was performed to provide a best estimate thermal-hydraulic and thermal-mechanical response of the fuel rod and capsule conditions. This coupling technique was previously reported in multiple works [7, 8]. Some post-test modelling results will be shown in the following section.

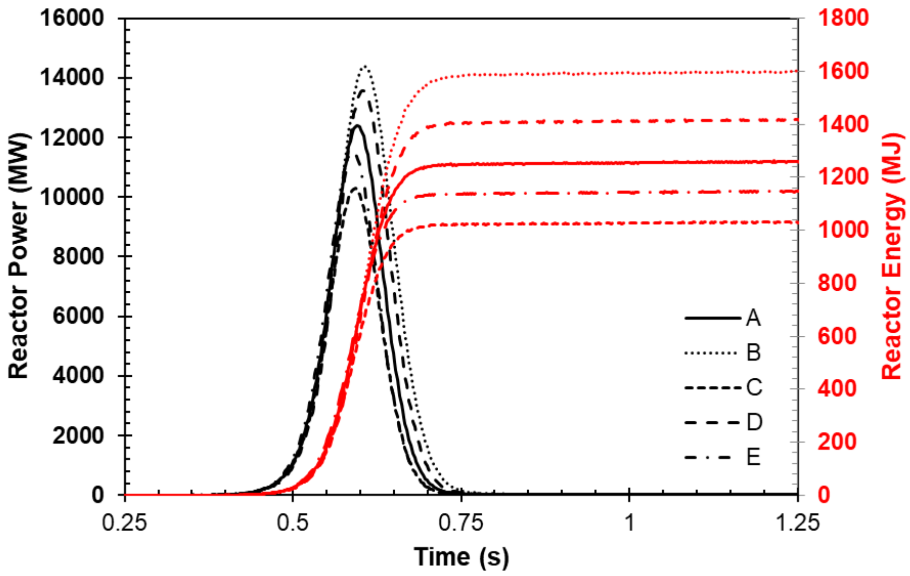

Fig 2. TREAT reactor power and energy deposition histories.

3. Results and Discussion

3.1 Calibration Test

The purpose of the calibration test was to verify the energy coupling factor between TREAT and the fuel specimen. This fuel contained both 4.95% and 0.74% U^{235} pellets and was irradiated with a small transient that deposited 100 MJ of energy in the TREAT core. Following the transient, gamma spectroscopy was performed on the fuel rodlet to determine the specimen coupling factor [9]. Fig 3 shows the results of the calibration test. This figure compares the measured results with their respective uncertainties from gamma spectroscopy to those predicted using MCNP. The predictions match up very well with the measured results, which gives confidence that the specimen heat generation rates predicted would provide an accurate representation to those in the experiments. The results show the very distinct change between the 4.95% and 0.74% fuel pellets. The values from pellets 2-4 and 7-9 were used to avoid errors/uncertainties due to end or transition effects.

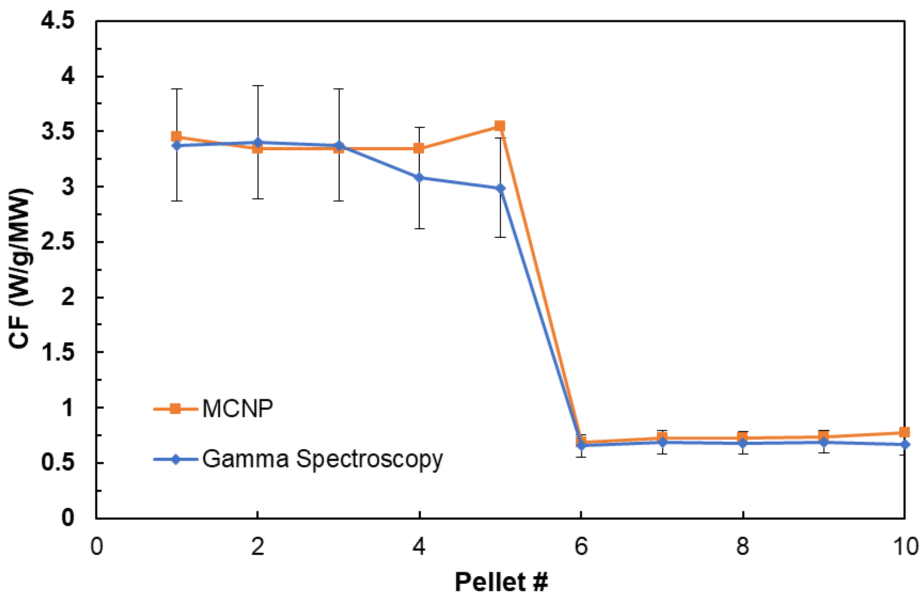

Fig 3. MARCH-SERTTA calibration test fuel coupling factor predictions vs. measurements

3.2 Temperature Predictions

Cladding and water thermocouple temperature results are plotted in Fig 4 for the ATF-RIA-1-E test along with the associated TREAT power. The analysis for Tests A–D is still ongoing, and results are not currently available for inclusion. The water thermocouple shows a small increase during the transient that is due to gamma heating in the water and thermocouple materials. The two cladding thermocouple measurements provide a good agreement on cladding temperature throughout the transient with slight deviations on the time of rewetting.

The data shown in Fig 4 is the raw data as collected by the data acquisition system, current efforts are in process to analyse this data in detail. The presence of thermocouples attached to the surface of the cladding have a localized impact on the measured temperature by the thermocouples. This impact can be over 100°C for fast transient such as these [10]. The data also shows a lot of oscillations that may be noise or actual temperature oscillations due to the presence of film boiling instabilities around the thermocouple to cladding weld. This has been observed in high-speed video of tests performed at INL using an out-of-pile pulsed power system that uses a bank of capacitors to drive DNB in tens of milliseconds on a rodlet-like heater rod [11].

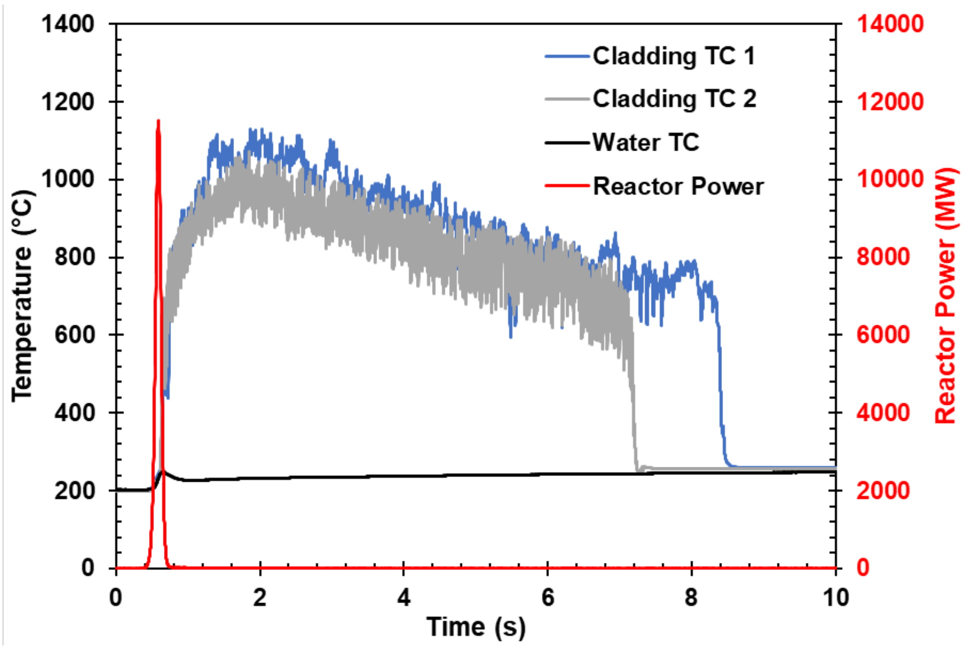

Fig 4. ATF-RIA-1-E cladding and water thermocouple measurements

One phenomenon that has been observed during fast temperature excursions, such as an RIA, is the shift in critical heat flux (CHF) to a higher value before the fuel rod experiences film boiling [12-14]. This topic has been the subject of recent research using the TREAT reactor at INL [15] and that phenomenon appears to be present in this test.

The measurements from the ATF-RIA-1-E test have been compared to both RELAP5-3D models and BISON/RELAP5-3D coupled models of the experiment. These results are shown in Fig 5. The thermocouple measurements plotted here are time-averaged over 0.15 seconds, as suggested in [16] for thermocouples of the size used in this experiment. The experimental results were compared with a number of different RELAP5-3D simulations with varied multiplication factors on parameters such as the CHF and the film boiling heat transfer coefficient. Varying the factor on the CHF changes the time when the boiling regime switches from nucleate boiling to film boiling, which is indicated by a rapid increase in cladding temperature. This can be seen at ~0.65 seconds in Fig 5b. The CHF factor also has a minor effect on the peak cladding temperature, since delaying DNB will result in more energy being transferred out of the fuel prior to DNB. The RELAP5-3D models show that a factor of 3.5 on the CHF provides very good agreement with the thermocouples, as well as the boiling detector data, on the time of DNB. This factor was used for all subsequent models shown in Fig 5.

The factor on the film boiling heat transfer coefficient has a significant impact on the peak cladding temperature reached after DNB. This can be seen by comparing the results in Fig 5a. The BISON/RELAP_0.75 designation indicates that the BISON and coupled RELAP5-3D model was used with a film boiling factor of 0.75. This resulted in a much higher peak cladding temperature of just over 1150°C as compared to the 1.5 film boiling factor with a peak cladding temperature of just over 1000°C, which showed very good agreement with the thermocouple data. The film boiling factors used in these models are just to demonstrate the variation that can be expected. Work to analyse these results more fully is currently ongoing, as the actual cladding temperatures could be much higher than the thermocouples measured as previously discussed.

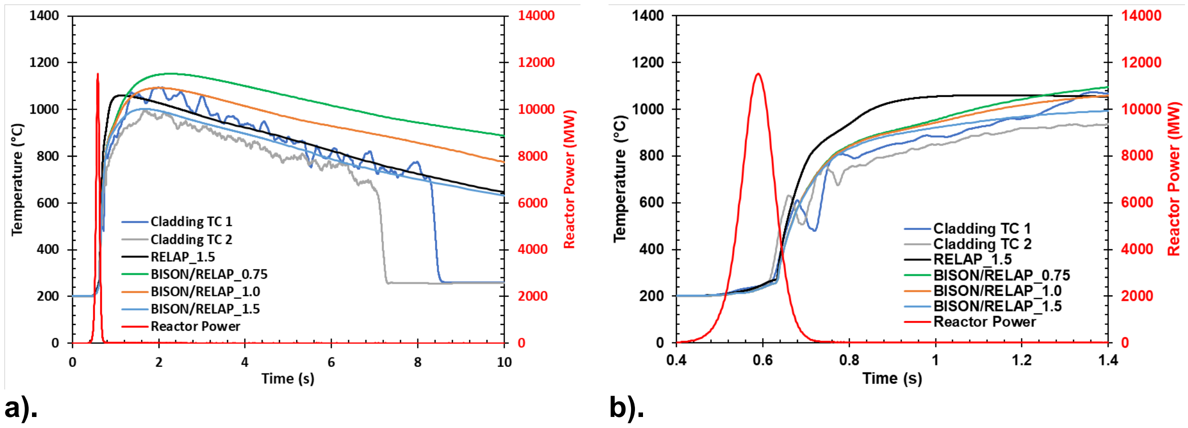

Fig 5. ATF-RIA-1-E cladding temperature measurements compared against RELAP5-3D and BISON/RELAP5-3D predictions

The RELAP5-3D and BISON/RELAP5-3D models showed very different predictions when it came to the time of cladding rewet than the thermocouple data. The thermocouples showed that rewet occurred around 8 seconds where the models predicted rewet anywhere from ~16–28 seconds. This is due to the localized fin effect the thermocouple has on the cladding temperature. This has also been observed in high-speed video, where the area around the thermocouple rewets considerably sooner than the rest of the rod. This can be seen from a still image from one such experiment shown in Fig 6, where the area around the thermocouple has rewet prior to the remaining rod because the presence of the thermocouple acts as a localized cooling fin. Additional work is needed to quantify the measurement impact of attaching thermocouples onto the surface of cladding.

3.3 Post-Irradiation Examination

Experiments A–D have undergone initial post-irradiation examination (PIE). X-ray radiography of the assembled capsules following the transient was performed. This provided a preliminary observation of the rodlet condition. An x-ray image of the ATF-RIA-1-B rodlet is shown in Fig 7. The image clearly shows the fuel pellets in the rodlet with the pellet gaps visible. The cladding shows irregular surfaces, indicating that a significant deformation of the cladding took place during this test. The test deposited 1,110 J/g (265 cal/g) into the fuel, which is above the failure limit even for fresh fuel [17, 18].

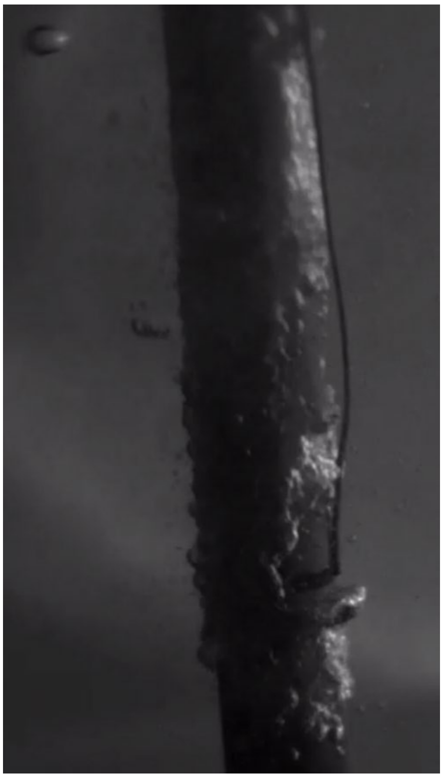

Fig 6. Still image from high-speed video showing rewet around thermocouple happening before rest of the rod

The MARCH-SERTTA capsules have been opened, and the rodlets extracted and visual inspections and diametral measurements have been performed for Rodlet A–D. Fig 8 shows images of the rodlets compared to similar historic tests that were performed under the SPERT-IV fresh fuel RIA tests program [19]. In all cases, the condition of the fuel rods shows good agreement with what has historically been observed. Rodlet B appeared to be intact during the x-ray radiography but upon visual inspection, the cladding was severely compromised, and the clear failure of the fuel rod is evident in the disassembled image. The remaining rods all stayed intact, and upon inspection, all the rods showed some degree of bowing. The bowing of the rodlets would be expected to occur due to deformation occurring at lower temperatures while the Zr cladding is in the hexagonal close-packed α-phase, which is anisotropic [20-24]. The location of the fuelled region is clear with a distinct oxide layer formed on the cladding. The PIE work is ongoing for all five rodlets (A–E), and further PIE including destructive examinations will be reported at a later time.

4. Conclusions

The completion of the ATF-RIA campaign concludes the commissioning tests for the water-based RIA testing in the MARCH-SERTTA experiment vehicle in TREAT. All tests were successfully completed with varied conditions from room temperature and pressure up to 200°C and 2.5 MPa, with energy depositions varying between ~500–1,100 J/gUO_2. Many of the test objectives were successfully reached. All tests experienced a DNB event, even when starting from room temperature conditions. Rodlet B experienced failure, as was expected during the transient. The tests also provided for a variety of different instrumentation studies, including LVDT-based plenum pressure and rodlet elongation studies. The tests provided valuable information on the performance of thermocouples during very fast transients, such as RIAs. The presence of thermocouples provides valuable temperature measurements, but there is ongoing work to understand the effect thermocouples have on the localized perturbations of actual cladding temperature.

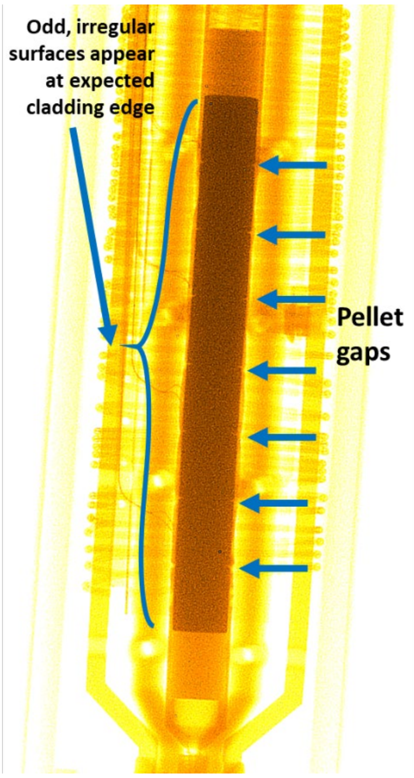

Fig 7. X-ray radiography image of the ATF-RIA-1-B rodlet. Image shows deformation of the cladding along the length of the fuelled section of the rodlet. Image courtesy of D. Chichester

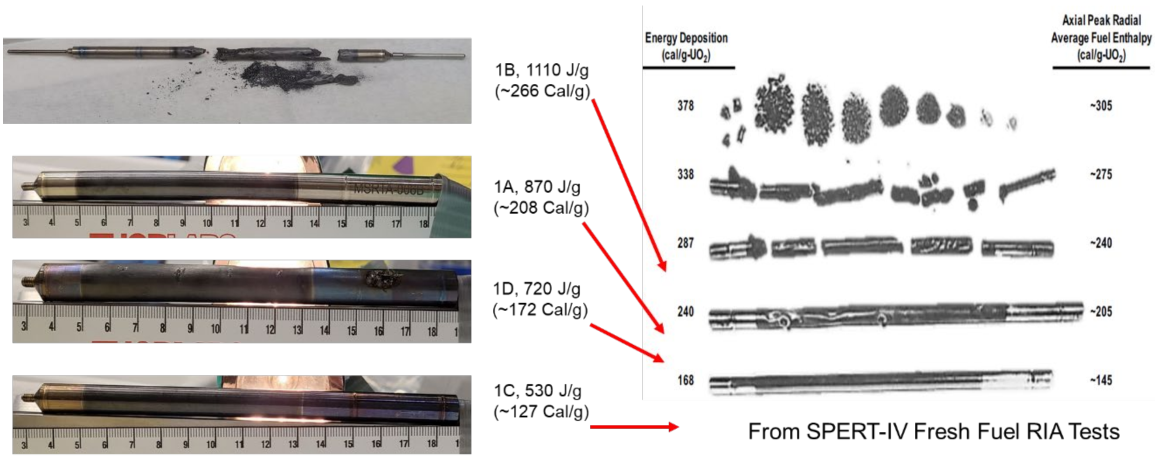

Fig 8. Visual inspections for Rodlets A–D compared to historic SPERT-IV fresh fuel RIA tests [19]

5. References

1. Kamerman, D.W., et al.,"Fuel Safety Research Plan for Accident Tolerant Fuels," INL/EXT-19-53343, 2019
2. Woolstenhulme, N., et al., "Development of Irradiation Test Devices for Transient Testing". Nuclear Technology, 2019. **205**(10): p. 1251-1265.
3. RELAP5-3D Code Manual Volume I: Code Structure, System Models and Solution Methods," INL/MIS-15-36723 Revision 4.3, 2015
4. Hales, J., et al., "Verification of the BISON fuel performance code". Annals of Nuclear Energy, 2014. **71**: p. 81-90.
5. Williamson, R., et al., "Validating the BISON fuel performance code to integral LWR experiments". Nuclear Engineering and Design, 2016. **301**: p. 232-244.

6. Williamson, R.L., et al., "BISON: A flexible code for advanced simulation of the performance of multiple nuclear fuel forms". Nuclear Technology, 2021: p. 1-27.
7. Folsom, C.P., et al., "BISON modeling of reactivity-initiated accident experiments in a static environment", in *Top Fuel 2016*. 2016, Idaho National Laboratory: Boise, Idaho.
8. Folsom, C.P., "Fuel Performance Modeling of Reactivity-Initiated Accidents Using the BISON Code". 2017, Utah State University.
9. Woolstenhulme, N., et al., "Core-to-specimen energy coupling results of the first modern fueled experiments in TREAT". Annals of Nuclear Energy, 2020. **140**: p. 107117.
10. Tsuruta, T. and T. Fujishiro, "Evaluation of thermocouple fin effect in cladding surface temperature measurement during film boiling". Journal of Nuclear Science and Technology, 1984. **21**(7): p. 515-527.
11. Terrill, K., et al., "Transient Boiling During RIA: Out-of-Pile Testing and In-Pile Design for TREAT", in *Global/Top Fuel 2019*. 2019: Seattle, WA.
12. Bessiron, V., T. Sugiyama, and T. Fuketa, "Clad-to-coolant heat transfer in NSRR experiments". Journal of nuclear science and technology, 2007. **44**(5): p. 723-732.
13. Bessiron, V., "Modelling of clad-to-coolant heat transfer for RIA applications". Journal of nuclear science and technology, 2007. **44**(2): p. 211-221.
14. Hernandez, R., et al., "Review of pool boiling critical heat flux (CHF) and heater rod design for CHF experiments in TREAT". Progress in Nuclear Energy, 2020. **123**: p. 103303.
15. Armstrong, R., et al., "Calculation of Critical Heat Flux Using an Inverse Heat Transfer Method to Support TREAT Experiment Analysis", in *28th Conference on Nuclear Engineering* 2020: Anaheim, California.
16. *Thermocouple Response Time*. 2019 [cited 2021 May 13]; Available from: https://www.omega.com/en-us/resources/thermocouples-response-time.
17. "Nuclear Fuel Safety Criteria Technical Review Second Edition," Nuclear Energy Agency No. 7072, 2012
18. "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," U.S. NRC Draft Regulatory Guide DG-1327, 2016
19. MacDonald, P., et al.,"Assessment of light water reactor fuel damage during a reactivity initiated accident," 1980
20. Chapman, R., et al., "Zirconium cladding deformation in a steam environment with transient heating", in *Zirconium in the Nuclear Industry*. 1979, ASTM International.
21. Abriata, J., J. Garces, and R. Versaci, "The O− Zr (oxygen-zirconium) system". Bulletin of Alloy Phase Diagrams, 1986. **7**(2): p. 116-124.
22. Dupin, N., et al., "A thermodynamic database for zirconium alloys". Journal of nuclear materials, 1999. **275**(3): p. 287-295.
23. Alam, T., et al., "A review on the clad failure studies". Nuclear Engineering and Design, 2011. **241**(9): p. 3658-3677.
24. Schulthess, J., et al., "Non-Destructive post-irradiation examination results of the first modern fueled experiments in TREAT". Journal of Nuclear Materials, 2020. **541**: p. 152442.