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Transient Irradiation Test Plan

Accident Tolerant Fuel Reactivity Initiated Accident Test Plan

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REVISION LOG

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1. PURPOSE AND BACKGROUND

The purpose of the Accident Tolerant Fuel-Reactivity Initiated Accident (ATF-RIA) transient irradiation tests (previously designated ATF-3-1 tests) is to understand ATF behavior when it is taken on a rapid high-power excursion typical of those seen in Light Water Reactor (LWR) design basis RIAs. These tests are one of three integral test series described in the "Fuel Safety Research Plan for Accident Tolerant Fuels." [1] Initially, the tests will be conducted with fresh fuel materials on designs that are representative of the ATF rods being irradiated as part of the lead test assembly (LTA) programs. Tests on fresh fuel will follow a systematic approach where the severity of the transient is gradually increased. In-situ instrumentation and post-transient examination will be used to identify fuel failure thresholds, as well as loss of coolable geometry thresholds for the fresh fuel concepts. Following the fresh fuel tests, previously irradiated rods will be tested at different burnup levels. Testing on irradiated materials will take place near the thresholds identified in the fresh fuel tests as well as relevant out of pile test data to identify any changes in behavior or failure thresholds as a result of the irradiation history.

In Pressurized Water Reactors (PWRs), the design basis RIA is the control rod ejection accident (CREA). In Boiling Water Reactors (BWRs), the design basis RIA is the control rod drop accident (CRDA). In both cases, the transient is most severe when the reactor is at a zero-power condition. CREAs and CRDAs result in the insertion of a prompt amount of reactivity, sending the reactor on a rapid positive power period. The transient is terminated as the fuel heats up, and Doppler broadening in the fuel decreases the excess reactivity. The resulting power pulse is nominally Gaussian in shape, the width of which can be characterized by a full width at half maximum (FWHM) or pulse width. For PWRs at hot coolant conditions and zero or low reactor power (such as the case just prior to startup) termed hot zero power (HZP), pulse widths range from 25 – 65 milliseconds (ms). For BWRs at HZP, the pulse widths range from 45 – 140 ms.

The purpose of this test plan is to describe the approach for assessing the behavior of ATF materials in simulated RIA accidents by conducting an integral transient testing campaign at the Transient Reactor Test (TREAT) facility at Idaho National Laboratory (INL). For RIA testing of ATF, a static water transient irradiation device named MARCH-SERTTA is being developed and commissioned. This test plan describes the irradiation experiments which will utilize TREAT and the MARCH-SERTTA environment to study the performance of LWR fuels in RIAs. It is anticipated that only minor changes to that experiment hardware will be required to execute this plan. INL Laboratory Wide Procedure (LWP)-20700 [2] describes the process for executing nuclear fuel irradiation experiments in nuclear test reactors. This plan is organized according to the principal tasks described in LWP-20700.

2. EXPERIMENT DESIGN AND ANALYSIS

As indicated in the introduction, it is anticipated that the ATF-RIA test series will take place in TREAT's MARCH-SERTTA Capsule shown below in Figure 1. The MARCH-SERTTA Capsule is a static water environment capable of some elevated temperature and pressure testing. Initial estimates for the pre-transient temperature and pressure capability of MARCH-SERTTA are around 200°C and 3 MPa. The ultimate pressure capability of MARCH-SERTTA is much higher. The capsule is equipped with an expansion chamber above the test specimen to allow for the rapid vaporization of the water during the transient, allowing for some testing of material performance following departure from nucleate boiling (DNB). A high temperature crucible is in place at the bottom of the capsule to allow for partial or complete melting of the test specimen.

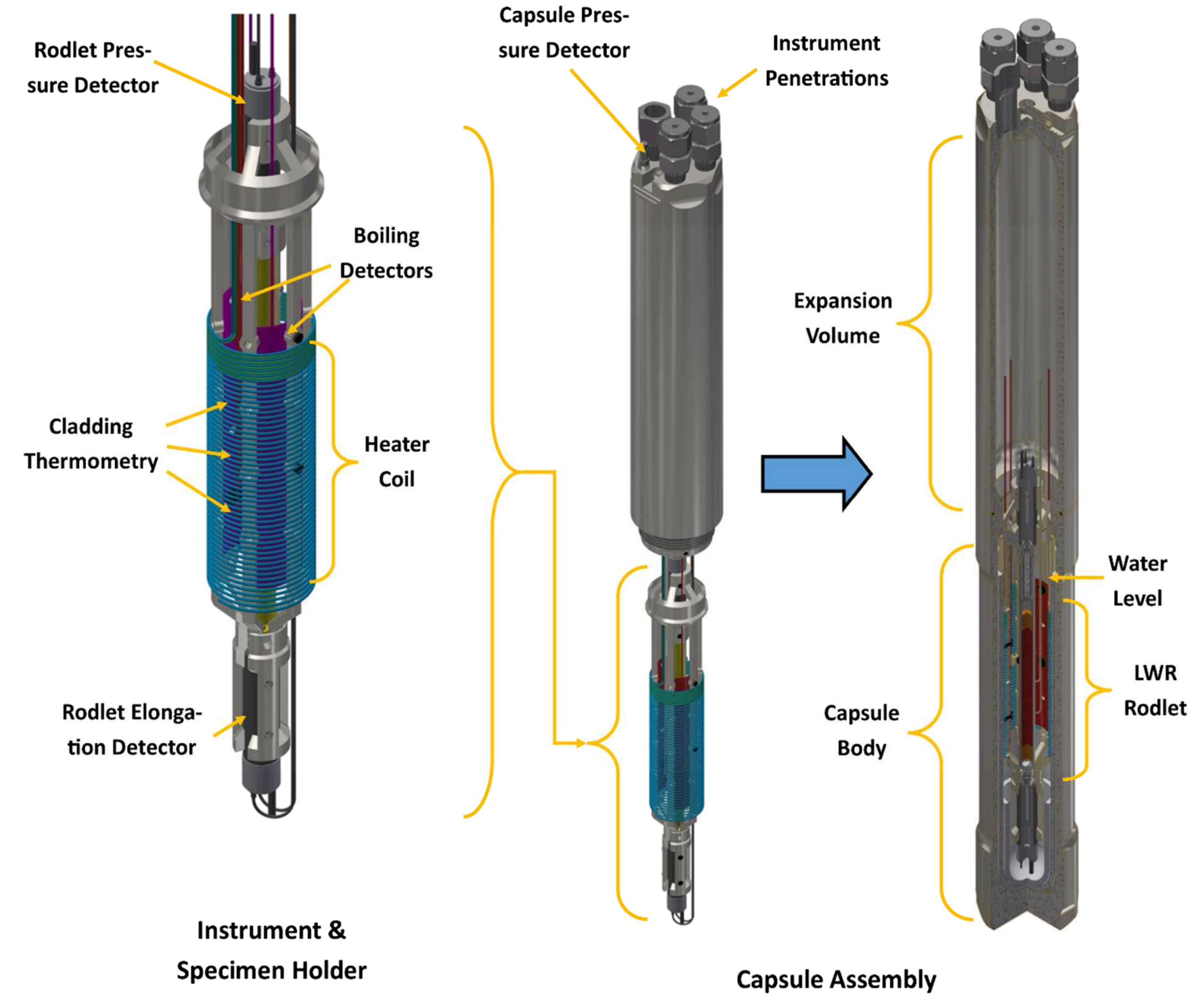

Figure 1. MARCH-SERTTA test train.

The test rodlet is anticipated to be ~15cm tall with ~9cm of active fuel height. The cladding outer diameter is expected to range from 9.4mm to 10.3mm. Internal diameter of the cladding and size of the fuel pellets are expected to vary based on the ATF design concepts for cladding thickness and pellet cladding gap. Rodlet internal pressure will also vary based on the design. Nominal internal pressures for PWR fuel rods is 2 MPa, while the internal pressure for BWR fuel rods is around 0.7 MPa. Twelve 1 mm instrument leads penetrate the capsule and can be used to accommodate a variety of different instruments. The baseline instrumentation package includes top and bottom linear variable differential transducers (LVDTs) to measure rod internal pressure and cladding axial expansion respectively. Capsule pressure will be monitored via the pressure transducer at the top of the capsule. A capacitive boiling detector will be used to measure the void fraction of the coolant surrounding the rodlet. Three thermocouples will be used to measure and control the heater temperature, 1 thermocouple will be used to measure the water temperature, and 4 thermocouples will be used to measure the cladding temperature. An optical pyrometer will also be used to measure the cladding temperature.

Only a fraction of the energy released by the TREAT reactor is deposited in the target fuel test specimen. The ratio of reactor power to specimen power is referred to as the power coupling factor (PCF). The coupling factor has units of W/g-MW. The parameter is strongly influenced by the neutronic characteristics of the test specimen and test capsule. The PCF will likely vary based on the initial enrichment and depletion characteristics of the fuel, the geometry of the fuel pin, temperature of the water, temperature of the fuel, and the void fraction. For test specimens in MARCH-SERTTA, the PCF values are expected to range from 0.3 to 3. Previously irradiated materials with higher burnup are expected to be towards the lower end of this range. Neutronics modeling is required to determine the effect environmental variables play in determining the PCF. An array of PCF values should be calculated when varying the enrichment from 0.711% to 4.9%, fuel pellet diameters from 8mm to 9mm in diameter, water temperature from 23°C to 200°C, and fuel temperatures from 23°C to 2900°C. Because the prompt RIA pulse will likely occur before any notable heat transfer to the capsule, the effect of coolant void formation does not need to be initially accounted for in the neutronic analysis. If notable phase change events are observed coincident with the nuclear transient, then the effects will be modeled and interpreted after the irradiation on an as-run basis.

The static water conditions of MARCH-SERTTA result in a pool boiling condition in the test rodlet. A thermal hydraulics model of the capsule is required to determine the effective heat transfer conditions on the cladding surface and the pressure inside the MARCH-SERTTA capsule. A coupled thermal mechanical finite element model of the test rod is required to determine the temperatures and stresses experienced by the rod during the transient. This analysis can use a commercial finite element code such as ABAQUS or INL's BISON fuel performance code. The combined thermal hydraulic and thermal mechanical modeling efforts should be to determine transition regions where given energy depositions result in cladding failure and/or fuel/cladding melting. The energy input should be an assumed Gaussian shape ranging from 500 J/g to 1300 J/g over a FWHM ranging from 70ms to 200ms.

3. EXPERIMENT TEST ROD FABRICATION

Test rod fabrication should follow an applicable Fabrication Control Plan for the test campaign. As-built dimensions and specifications should be well documented and incorporated back into the analysis tasks described above. Important as-built specifications of the test rodlet include:

- Pellet Height
- Pellet Material (UO_2, U_3Si_2 etc.)
- Notable Dopant or Impurity Concentrations
- Notable Stoichiometry deviations
- Number of Pellets in Stack
- Pellet Diameter
- Pellet Dish and Chamfer Dimensions.
- Pellet Enrichment
- Pellet Density and Mass
- Total Fuel Rod Length
- Cladding Outer Diameter
- Cladding Inner Diameter
- Cladding Material (Zr-4, M5^{TM}, C26M, SiC-SiC)
- Composition and Thickness of any Cladding Coatings or Liners (Cr, CrN, SiC)
- Height and Width of any Spacers
- Mass of any Spacers
- End Cap Dimensions
- Internal Instrumentation Dimensions (Push Rod or Bellows)
- Internal Gas Volume
- Internal Gas Pressure
- Internal Gas Composition
- Total Mass of the as-built Rodlet
- Location of Attached Rodlet TCs.

For previously irradiated fuel rods, additional information is required, most of which will come from the Pre-Transient Material Characterization step described in the next section. Noncommercial feedstocks should have a full chemical analysis performed to supply identified levels of chemical impurities. Previously irradiated rods will require special techniques to be developed for their re-fabrication and insertion into the MARCH-SERTTA test train. Re-fabrication includes incorporating any instrumentation specified in the design and hermetically sealing the rod to the specified internal pressure.

4. PRE-TRANSIENT MATERIAL CHARACTERIZATION

Pre-Transient Characterization activities will be centered around determination of the mechanical and thermophysical properties for the fuel and cladding materials. For commercially supplied feedstocks, literature values can be used and little to no pre-transient material characterization needs to be performed. For novel fuel or cladding products as well as previously irradiated materials, Pre-Transient Material Characterization shall be performed. In steady state irradiations, examination of irradiated materials normally takes place in the Post Irradiation Examination (PIE) phase of the test. However, for transient tests of previously irradiated material, these kinds of examinations are required prior to the irradiation test and are discussed here in the characterization phase.

Determination of the test material's thermal mechanical state as well as the extent of environmental degradation is important to know prior to specifying the transient conditions, so that an accurate understanding of the cause of any burnup dependent failure mode can be determined. Some of the characterization activities are destructive in nature and require two test rods, one for the transient test, and one for destructive examination herein referred to as the sister rod. The two rods could be from adjacent positions in the ATR or could be from the same test rod in an LTR at similar axial locations. Details of the specific exams requested will be prepared in a separate ATF-RIA Pre-Transient Material Characterization Report. Important characteristics of the fuel rod are included in a bulleted list below:

- Cladding oxide and crud layer thickness
- Cladding diameter as a function of axial position
- Cladding excess hydrogen content and hydrogen morphology
- Nature and extent of any cladding defects
- Internal pressure
- Amount of fission gas in plenum (gap inventory)
- State of pellet cladding gap
- Burnup and Isotopic inventory of fuel
- Mechanical testing of cladding.

Mechanical testing of the cladding samples will be conducted both in the hoop and axial directions. The mechanical test samples will be milled out of the cladding samples. The sample geometry will consist of axial tensile specimens and ring stretch specimens described in "Mechanical Property Testing of Irradiated Zircaloy Cladding Under Reactor Transient Conditions." [3] Mechanical testing will take place at room temperature and at 200°C which is the approximate maximum pre-transient temperature of the MARCH-SERTTA irradiation vehicle. Following characterization of the test rod and sister rod, the test rod will need to be refabricated prior to the transient test.

5. TRANSIENT IRRADIATION TESTING CONDITIONS

The overall objective of the ATF-RIA test series is to identify thresholds related to fuel rod damage, cladding failure, and the loss of fuel rod coolable geometry for prototypic LWR fuel designs in design basis RIA events. The test matrix involves subjecting test rodlets in increasingly severe RIA-like transients at TREAT's narrowest pulse width of around ~90ms. When necessary to deposit sufficient energy, large pulse widths up to 200ms can be used. The transient clipping system should generally be employed to limit additional energy release after the Gaussian phase, with the exception of transients where significant fuel disruption is expected. In those transients, fission energy should be supplied after the pulse sufficient to observe fuel motion in the fast neutron hodoscope. The energy deposition targets supplied in the test matrix are assumed to be Gaussian in their deposition shape, where the Gaussian energy is defined as:

$$Gaussian\ Energy = Peak\ Power * FWHM * \sqrt{\frac{2\pi}{2.35}} \tag{1}$$

The ATF-RIA experiment test series is currently planned with three primary phases. The first phase will be with Urania pellets and Zircaloy cladding, and the second phase will incorporate the various ATF materials. The third phase will include previously irradiated materials. The goal of the first phase is to observe predicted behavior consistent with expected failure modes. The goal of the second phase is to explore the behavior of the ATF materials in RIA transients. Finally, the goal of the third phase is to assess burnup dependence on the failure modes. Two temperature levels are prescribed, a room temperature condition estimated at 30°C, and a high temperature condition which will be the maximum operating temperature of the MARCH-SERTTA Capsule currently estimated at 200°C. The capsule will be pressurized to provide an equivalent level of sub-cooling relative to a PWR or BWR at HZP or cold zero power (CZP) conditions.

5.1 ATF-RIA Phase 1 Testing Matrix

The test matrix for Phase 1 of the ATF-RIA tests is presented in Table 1 below. The goal of the test is to replicate a prototypic PWR rodlet design in an increasingly severe HZP CRDA.

Table 1. ATF-RIA Phase 1 test matrix.

Test Rodlet Number	Clad Material	Pellet Material	Rodlet Pressure	Capsule Temperature	Gaussian Energy Release Target
ATF-RIA-1-A	Zircaloy	UO2	2 MPa	30°C	700 J/g
ATF-RIA-1-B	Zircaloy	UO2	2 MPa	30°C	1000 J/g
ATF-RIA-1-C	Zircaloy	UO2	2 MPa	200°C	700 J/g
ATF-RIA-1-D	Zircaloy	UO2	2 MPa	200°C	1000 J/g
ATF-RIA-1-E	Zircaloy	UO2	2 MPa	200°C	1300 J/g

5.2 ATF-RIA Phase 2 Testing Matrix

The Phase 2 test matrix is present in Table 2 and consists of 2 different combinations of ATF materials which are representative of near-term and longer-term concepts. The uranium silicide tests have notably lower energy-deposition targets, allowing for this material's notably lower heat capacity and melting point from UO_2. These enthalpy targets may be adjusted, based on the results of upcoming transient tests on U_3Si_2 in the ATF-SETH testing campaign.

Table 2. ATF-RIA Phase 2 test matrix.

Test Rodlet Number	Clad Material	Pellet Material	Rodlet Pressure	Capsule Temperature	Gaussian Energy Release Target
ATF-RIA-2-A	Coated Zircaloy	Doped UO2	2 MPa	30°C	700 J/g
ATF-RIA-2-B	Coated Zircaloy	Doped UO2	2 MPa	30°C	1000 J/g
ATF-RIA-2-C	Coated Zircaloy	Doped UO2	2 MPa	200°C	700 J/g
ATF-RIA-2-D	Coated Zircaloy	Doped UO2	2 MPa	200°C	1000 J/g
ATF-RIA-2-E	Coated Zircaloy	Doped UO2	2 MPa	200°C	1300 J/g
ATF-RIA-3-A	SiC-SiC	U3Si2	2 MPa	30°C	300 J/g
ATF-RIA-3-A	SiC-SiC	U3Si2	2 MPa	30°C	350 J/g
ATF-RIA-3-B	SiC-SiC	U3Si2	2 MPa	200°C	300 J/g
ATF-RIA-3-C	SiC-SiC	U3Si2	2 MPa	200°C	350 J/g
ATF-RIA-3-D	SiC-SiC	U3Si2	2 MPa	200°C	450 J/g

5.3 ATF-RIA Phase 3 Testing Matrix

An initial test matrix for previously irradiated materials is presented in Table 3. These three transients are not meant to be exhaustive of the RIA testing that will take place on previously irradiated materials and is expected to form the bulk of the ATF-RIA testing in the upcoming years. The tests are labeled starting with ATF-RIA-7. Test designators ATF-RIA-4,-5,-6 are left open for the addition of any rodlets based on industry-sponsored testing.

The test matrix presented is a starting point for demonstrating the ability to test previously irradiated material. Test conditions will be refined based on pre-transient characterization activities and set at the current regulatory limits for fuel failure (high and low temperature based on cladding hydrogen concentration and internal gas pressure) and coolable geometry. Internal pressure and capsule temperature will be used in the refabrication step to promote either a high temperature or pellet-to-cladding mechanical interaction (PCMI) (low temperature) failure mode.

Table 3. ATF-RIA Phase 3 test matrix.

Test Rodlet Number	Test Material	Internal Pressure	Capsule Temperature	Radial Average Enthalpy Target
ATF-RIA-7-A	High Burnup UO_2-Zircaloy	0.1 MPa	30°C	630 J/g
ATF-RIA-7-B	High Burnup UO_2-Zircaloy	4-6 MPa	200°C	710 J/g
ATF-RIA-7-C	High Burnup UO_2-Zircaloy	2 MPa	200°C	960 J/g

6. POST-TRANSIENT EXAMINATION

Post-Transient Irradiation Examination will take place to determine the damage state and damage mechanisms of the test rodlet. A detailed Post-Transient Irradiation Examination Plan will be written to provide detailed instructions to the facilities involved. A brief overview of the requested exams is provided herein for reference.

A standard suite of non-destructive examinations shall take place on the samples after the RIA transient. A detailed visual examination of any cladding ruptures should take place, including characterization of the length and area of any cladding ruptures. Axial profilometry should take place at a minimum of two azimuthal orientations in order to determine the final diametral strain of the cladding.

A sectioning plan for the test rodlet should be developed to ensure the maximum amount of data from destructive microstructural and mechanical evaluations can be obtained. Radial sections should be made to optically observe crack patterns in the fuel pellet. Additionally, any regions of partial melting should be examined to determine the extent of any chemical interaction. Detailed microstructural examinations should take place on the cladding to examine the crack pattern from any failed rodlets and to evaluate any chemical or microstructural changes (such as changes in hydride morphology and distribution and/or oxidation behavior) across the cladding thickness as a result of the high temperature excursion. A chemical analysis of the cladding should be done to confirm the hydrogen concentration. Exams should be done both near any failed regions in the cladding and away from any failed regions of the cladding. In addition to the axial tension and ring stretch mechanical tests described in the characterization section, ring compression tests consistent with Nuclear Regulatory Commission (NRC) document DG-1262 [4] should be conducted at room temperature and at 135 °C to measure the amount of residual ductility in the cladding following the high temperature transient. Ring compression samples should be taken from regions near any potential failed region and far from failed regions.

7. REFERENCES

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[4] Nuclear Regulatory Comission, “Draft Regulatory Guide DG-1262 Testing for Post Quench Ductility,” 2014.