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Revision 0

Fuel Safety Research Plan for Accident Tolerant Fuels

Nuclear Technology Research and Development

Prepared for
U.S. Department of Energy
Advanced Fuels Campaign

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SUMMARY

This document outlines a strategic plan for a safety research program for accident tolerant fuels. Accident tolerant fuels (ATFs) are nuclear fuel materials and designs for existing and future light water reactors. This effort is under the Department of Energy Office of Nuclear Energy's Nuclear Technology Research and Development Advanced Fuels Campaign. The development of ATF is an industry lead initiative with fuel designs being championed by Westinghouse, Framatome, and General Electric. Industry goals for ATF deployment include implementing the first batch reloads of coated cladding and doped fuel designs into commercial reactors by 2023 and having the first batch reloads of more advanced designs into commercial reactors by 2026. Follow on goals include implementing the first full cores of coated cladding and doped fuel designs into commercial reactors by 2026 and implementing full cores of more advanced designs by 2030. The objectives of the national laboratory research programs within the Advanced Fuels Campaign related to ATF are:

1. Develop and maintain of unique nuclear testing infrastructure, which includes test reactors, hot cell PIE equipment, and unique out-of-pile separate effects testing facilities.
2. Respond directly to industry requests for support by partnering with the ATF vendors and executing experiments and/or analysis on their behalf.
3. Conduct independent scientific investigations into the performance of the different ATF concepts that are relevant to Industry designs.

To achieve these objectives, the safety research program will execute transient irradiation experiments at the TREAT facility at Idaho National Laboratory with the aim of identifying the relevant transient fuel behaviors for each fuel design in order to establish appropriate fuel safety criteria and thresholds. Two test series are described in this plan, (1) an Integral Test Series, and (2) a Focused Effects Test Series. In fulfillment of the first objective for the safety research program, Integral Effects Tests will be developed and demonstrated using standard UO_2-Zircaloy fuel designs in both the fresh and previously irradiated case. The Integral Test Series will consist of three campaigns to evaluate transient fuel performance in Reactivity Initiated Accidents, Loss of Coolant Accidents, and in generalized Power Cooling Mismatch scenarios. Development of these test programs also enables the safety research program to fulfill its second objective by making these three test programs available to each industry team. Initially, priority will be given in the integral test series to the coated cladding and doped fuel concepts as their deployment schedule is the most aggressive.

To fulfill the third objective of the safety research program, a Focused Effects Test Series is also being developed which will examine specific transient fuel behaviors that are of general interest to all the ATF designs under development. These tests look at transient fuel behavior related to pellet cladding mechanical interaction, coated cladding swelling, rupture, and time at temperature behavior, non-oxide fuel melting and mechanical interaction with Silicon Carbide cladding, UO_2 and doped UO_2 fuel fragmentation and transient fission gas release, and finally fuel coolant interaction.

This document provides a general overview of the objectives of the safety research program as well as providing background into how programs like this have been used in the past to develop the criteria necessary for the safe operation of the present fuel designs. An overview of the materials and methods used for safety testing is presented. Finally the Integral and Focused Effects tests series are introduced and described. This document will be maintained as a "living" document with annual updates to reflect evolving requirements and needs.

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FUEL SAFETY RESEARCH PLAN FOR ACCIDENT TOLERANT FUELS

1 INTRODUCTION

The establishment of a fuel safety research program for accident tolerant fuels (ATF) is essential to meet the Department of Energy (DOE) Office of Nuclear Energy (NE) needs under the Accident Tolerant Fuel (ATF) program, especially at an opportune time when the Transient Reactor Test (TREAT) facility is again operational at Idaho National Laboratory (INL). DOE began pursuing the development of light water reactor (LWR) fuels with enhanced accident tolerance in response to the 2011 earthquake and tsunami which lead to core damage at the Fukushima Daiichi Nuclear Power Stations in the Fukushima prefecture of Japan. The goal of the ATF program is to develop the next generation of LWR fuels which have significantly improved performance in design basis and beyond design basis accidents (DBAs and BDBAs)[1]. The fuel safety research program will aid the development of ATF concepts by developing robust data sets describing key transient fuel behaviors which enable the development of nuclear fuel safety criteria and support establishment of analytical limits applicable to ATF designs.

The ATF program is an industry lead program within DOE. As such, selection of the specific technologies (materials and designs) being pursued is decided upon by the three nuclear fuel vendors in the United States which include Westinghouse, General Electric, and Framatome (formerly AREVA). Each fuel vendor receives funding directly by DOE through cooperative agreements issued as part of a funding opportunity announcement (FOA) to pursue the ATF designs of their choosing. It is ultimately the responsibility of each fuel vendor to develop the data necessary to support the licensing and qualification of their specific ATF concepts. The ATF program had an initial goal of inserting ATF concepts as lead test assemblies (LTAs) into commercial reactors by 2022. However, due to industry demand for acceleration of the program, LTAs will have been inserted by each vendor team into commercial reactors by the conclusion of 2019. Thus, the ATF program has developed two follow-on goals. The first is to begin batch reloads of ATF by 2026 and the second is to have full cores of ATF by 2030. However, industry is hoping to accelerate these dates for some near-term ATF concepts, such as coated claddings and doped fuel pellets by having batch reloads available in 2023 and full cores of ATF operating in 2026 [2]. The national laboratory role within the ATF program is a support role which includes three major objectives:

1. Develop and maintain unique nuclear testing infrastructure, which includes test reactors, hot cell PIE equipment, and unique out-of-pile separate effects testing facilities.
2. Respond directly to industry requests for support by partnering with the ATF vendors and executing experiments and/or analysis on their behalf.
3. Conduct independent scientific investigations into the performance of the different ATF concepts that are relevant to Industry designs.

The national laboratory ATF research budget comes from the Advanced Fuels Campaign (AFC) within the Nuclear Technology Research and Development (NTRD) Program. The budget that is supplied directly to INL from the AFC will be used to accomplish objectives (1) and (3) above. To accomplish objective (2), INL will partner with each industry team and execute fuel safety research for their specific ATF concept. Funding for this work will come from the individual vendors FOA budgets.

1.1 Purpose and Objectives

The purpose of the fuel safety research program is to conduct scientific investigations into the performance of ATF concepts in transient conditions. These include anticipated operational occurrences (AOOs), DBAs, and BDBAs. The objectives are to replicate events, observe unique transient fuel behaviors, develop fuel safety criteria addressing these behaviors, and define analytical limits at identified thresholds.

The most fundamental idea in the development of nuclear fuel safety criteria is that the consequences of postulated events (transients) should be inversely proportional to the probability of those events. During events with higher probabilities typical of operating nuclear power plants, such as AOOs, fuel safety criteria generally allow for no damage to the fuel system. For DBAs fuel failure is generally an expected part of the accident sequence, and fuel safety limits are designed to ensure that the nuclear fuel maintains a geometry that is amenable to long term cooling and that radiological releases from failed fuel elements are quantified [3]. Limits in DBAs, therefore, are established to preclude events such as coolant channel blockage, fuel rod fragmentation, cladding embrittlement, and/or fuel rod melting. In DBAs, it is also generally required to quantify the number of fuel rod failures and estimate the release of volatile fission products to the reactor environment. This requires development of thresholds related to transient fission gas release as well as fuel fragmentation relocation and dispersal. In BDBA, it is only necessary to quantify the release fraction from the event as such events are by definition beyond the design capabilities of the reactor system. Figure 1 breaks down how this general safety philosophy flows down to the specific fuel safety criteria that are developed for LWR materials.

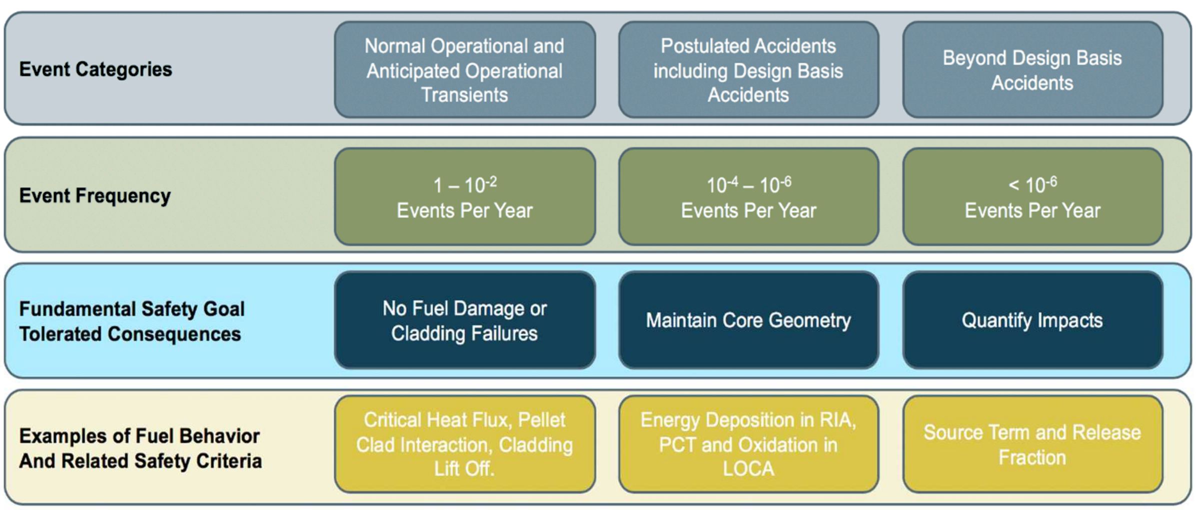

Figure 1. Generalized philosophy for development of fuel safety criteria.

1.2 R&D Approach

The approach followed by the fuel safety research program centers around the execution of specific transient fuel irradiation experiments executed at the TREAT facility at INL. If recommendations made by the INL to install additional testing loops at the Advanced Test Reactor (ATR) are accepted by DOE additional irradiation experiments will be added to the program which will make use of that testing infrastructure [4]. The transient testing program will be split into a series of three integral transient testing campaigns and five focused effects testing campaigns. The integral ATF testing campaigns will focus on creating testing conditions that are prototypic of those likely to be found in LWR transients. The fuel safety research program will develop the capabilities to conduct these tests and demonstrate them with both fresh and high burnup UO_2-Zircaloy materials using AFC program funding. Testing on specific fuel vendor's concepts, both the fresh fuel and previously irradiated ATF materials will be funded with the fuel vendors FOA budget. The three integral testing campaigns will focus on replication of design basis Reactivity Initiated Accidents (RIAs), Loss of Coolant Accidents (LOCAs), and generalized power cooling mismatch (PCM) scenarios. A description of this test program is included in Section 4 below. The focused effects testing campaigns will be more focused on quantifying specific transient fuel behaviors from a generic or technology neutral point of view. These tests look at transient fuel behavior related to pellet cladding mechanical interaction (PCMI), coated cladding swelling, rupture, and time at temperature behavior, non-oxide fuel melting and mechanical interaction with Silicon Carbide cladding, UO_2 and doped UO_2 fuel fragmentation and transient fission gas release, and finally fuel coolant interaction (FCI). A description of this testing program is included in Section 5 below.

The in-pile irradiation experiments will be supported by synergistic out of pile experiments, advanced modeling and simulation activities, and the development of unique In-situ In-pile instrumentation. The execution of irradiation experiments at INL follows INL's conduct of research principals and procedures [5]. These can be summarized into three phases of planning, execution, and communication. Planning and execution of irradiation experiments follows INL LWP-20700 [5] which defines the work breakdown structure (WBS) of irradiation experiments as (1) manage, (2) design, (3) fabricate, (4) characterize, (5) irradiate, (6) post irradiation examination (PIE). The out of pile, modeling and simulation, and instrumentation development activities that support the irradiation experiments will be included in the one of these six WBS elements which they support. This will keep the focus of these activities on supporting the in-pile experiments.

Communication will be achieved through a variety of means including technical reports, scientific publications, and through INL's Nuclear Data Management System (NDMAS). Technical reports will be written to DOE and to sponsoring industry partners describing the materials, methods, results, and conclusions of the transient irradiation experiments and supporting activities on regular intervals. Reports may be restricted in their distribution as they may contain proprietary information for each fuel vendor's ATF design. In addition to technical reports, fully open scientific publications will be generated by the INL staff executing the experiments. Scientific publications may take on a variety of forms including conference papers, journal articles, as well as invited presentations at universities, professional society meetings, national, and international forums. In addition, data from the irradiation experiments will be captured and stored in INL's NDMAS [6]. The database will include restricted and unrestricted web-based portals for stakeholders to access the data that are being generated from the irradiation experiments.

2 THEORY AND BACKGROUND

The study of the transient behaviors of nuclear reactor fuels requires an environment which can replicate the unique nuclear, thermal, and environmental conditions in which these events take place. While complete, prototypical replication of all three conditions is unlikely in any experiment, transient irradiation experiments can provide a unique replication of many specific aspects of these environments. As such, transient irradiation experiments are key to the development of the nuclear fuel safety criteria needed to license a new nuclear fuel design.

The development of nuclear fuel safety criteria and associated analytical limits is an evolutionary process. Sufficient data must be generated to support the initial licensing of a given fuel concept. However, data gaps may require that conservative limits be placed on the operations of new nuclear fuel concepts until such time as sufficient data are generated to extend their operating envelope. The development of safety criteria for the use of uranium dioxide fuel pellets encased in a Zircaloy cladding has a history almost as long as the nuclear power industry itself. Criteria and limits must remain adaptable to change as credible results from nuclear fuel safety research activities are published. Section 2.1 below described the general requirements for the development of safety criteria and limits from applicable Nuclear Regulatory Commission (NRC) regulations. Sections 2.2, 2.3, and 2.4 describe the transient fuel behavior and nuclear fuel safety criteria for UO_2-Zircaloy fuel designs in two specific categories of transients; RIAs and LOCAs, and in generalized PCM scenarios.

2.1 General Requirements for Nuclear Fuel Safety Criteria

Fuel safety criteria are developed to ensure that requirements of nuclear power plant (NPP) design and operations are met. The NRC standard review plan (SRP) of the nuclear fuel system design found in NUREG 0800 chapter 4.2 [7] establishes four objectives for fuel safety criteria which are to ensure that:

1. The nuclear fuel system is not damaged as part of normal operations including AOOs.
2. The fuel system damage is never so severe as to prevent control rod insertion when it is required.
3. The number of fuel rod failures is not underestimated.
4. Coolability is always maintained.

The first objective that the fuel system is not damaged as the result of normal operations and AOOs stems from General Design Criterion (GDC) 10, within Appendix A to 10 CFR Part 50 which requires that the reactors be designed to prevent fuel damage in both normal operations and AOOs. In this context "not damaged" means that the fuel rods do not fail, that the fuel system dimensions remain within operational tolerances, and that the functional capabilities of the fuel system are not reduced below those assumed in the safety analysis. Objectives (2) and (4) stem from requirements in 10 CFR Part 50 Appendix A, GDC 27 and 35 for the reactivity control system (RCS) and emergency core cooling system (ECCS) respectively. GDC 27 requires that the RCS be designed to be capable of controlling reactivity in postulated accidents. GDC 35 requires that the ECCS be capable of transferring heat from the reactor core following a loss of active cooling. In this context, "coolability" means that the fuel assembly retains a rod-bundle like geometry with adequate cooling channels to permit removal of residual heat. Objective (3) stems from requirements in 10 CFR Part 50.67 and Part 100 which require dose consequence analysis for all DBAs. Thus, the release fraction from failed fuel must be accounted for in those scenarios. In this context, "fuel failure" is defined as a loss in fuel rod hermeticity or fuel rod cladding breach. It is worth noting that it is not required that all fuel failures be prevented, even in normal operations. Indeed, the presence of cleanup systems in nuclear power plant designs presume that a small number of leaking rods may exist from time to time. However, in order to meet the requirements of GDC 10 (Objective 1) and to support dose consequence analysis as part of

DBAs (Objective 3), the SRP states that fuel failure criteria should be provided for all known fuel rod failure mechanisms. These criteria are used to demonstrate the “no damage” requirement in normal operations and AOOs and to meet the dose-consequence requirement in DBAs.

In order to meet these four objectives, fuel safety criteria are identified at three levels. These levels include (1) fuel damage criteria, (2) fuel failure criteria, and (3) fuel coolability criteria. While the criteria in the SRP are acceptable to meet the requirements of NRC regulations, they are not a substitute for the regulations, and compliance with them is not explicitly required. Changes in the fuel design, including changes to dimensions and/or materials, require a review to determine whether the SRP criteria are still applicable and if additional criteria need to be developed. For ATF designs, such a review will be required, and transient testing will aid the development of applicable criteria to those designs.

Nuclear fuel safety criteria are often grouped together to define the design and operational limits for the nuclear fuel system. Figure 2 below illustrates how nuclear fuel safety criteria for fuel damage and fuel failure are combined to establish an operations envelope consisting of a linear heat generation rate (LHGR) limit, and burnup limit. If overly conservative criteria are applied due to the lack of test data, it will limit the operational capability of the new fuels. Likewise, if criteria are not developed for new fuel designs because of unidentified phenomenon, then operation of that fuel may result in a violation of NRC safety requirements.

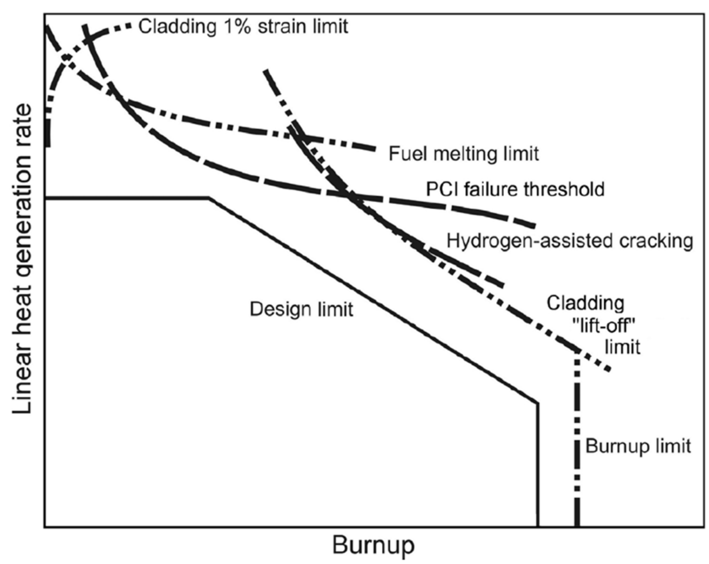

Figure 2. LHGR as a function of burn-up derived from Fuel Safety Criteria [3].

2.2 Transient Behavior and Fuel Safety Criteria for Reactivity Initiated Accidents

Reactivity initiated accidents (RIA) are transients that occur as the result of a sudden increase in the reactor’s reactivity. 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 accident 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 prompt part of the transient is followed by a decaying exponential tail which is the result of delayed neutron fissions. The point which divides the prompt Gaussian phase from the later decaying exponential phase of the transient is generally taken as one FWHM past peak power.

The study of LWR fuel performance in RIAs has a long history going back to the beginnings of LWR technology [8]. The understanding of fuel performance during RIAs, the associated safety criteria, and the testing approach have all evolved and matured through time. The key physical phenomena at play in an RIA transient is illustrated graphically in Figure 3. While a general overview is provided in the following paragraphs, an in-depth discussion on these physical phenomena is beyond the scope of this test plan, and the reader is referred to a number of other sources for more information on the specifics. Recommended general overviews of RIAs from a fuel performance perspective include:

- "Nuclear Fuel Behavior Under Reactivity Initiated Accident (RIA) Conditions" [9]
- "Transient Response of LWR Fuels (RIA)" [10]
- "Phenomenon Identification and Ranking Tables for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel" [11].

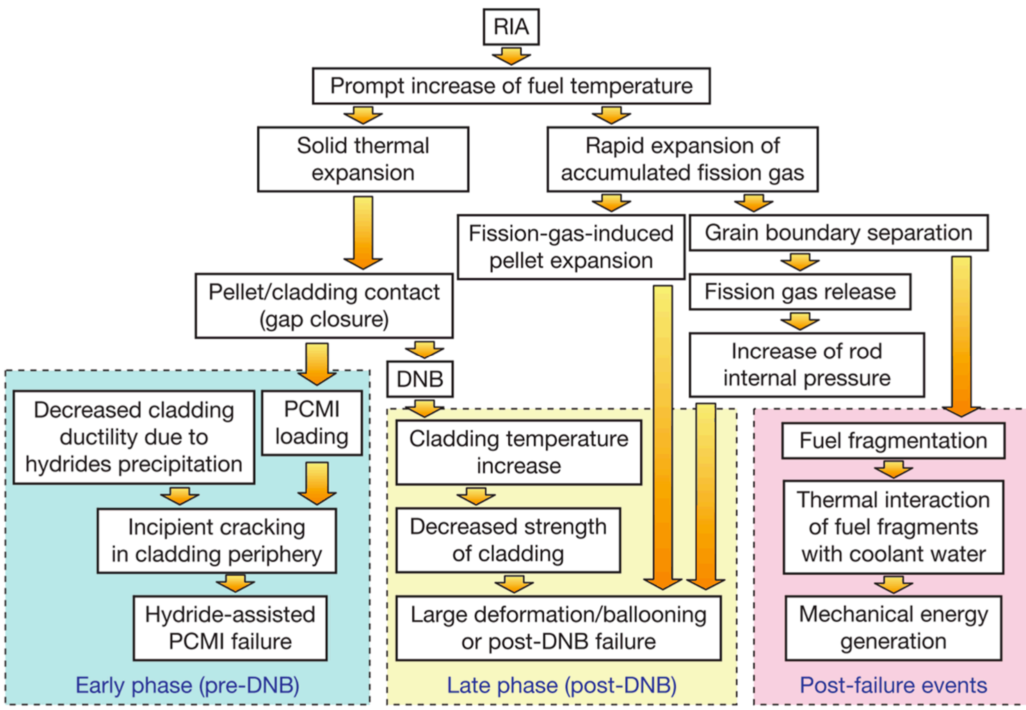

Figure 3. Transient Fuel Behavior in an RIA [10].

In LWR design basis RIAs, the rate of energy injection in the fuel can be much greater than the thermal response time of the fuel. Initially, the heatup in the fuel is so rapid that it happens nearly adiabatically with little sensitivity to the fuel's thermal conductivity or the cladding coolant heat transfer coefficient. In this early part of the transient, the fuel's enthalpy rise is nearly equivalent to the energy deposited in the fuel. However, as the power level in the transient begins to decrease and heat begins to be transferred from the fuel to the environment, the enthalpy increase in the fuel begins to diverge from the total energy deposited and, eventually, the fuel enthalpy begins to decrease, as illustrated in Figure 4.

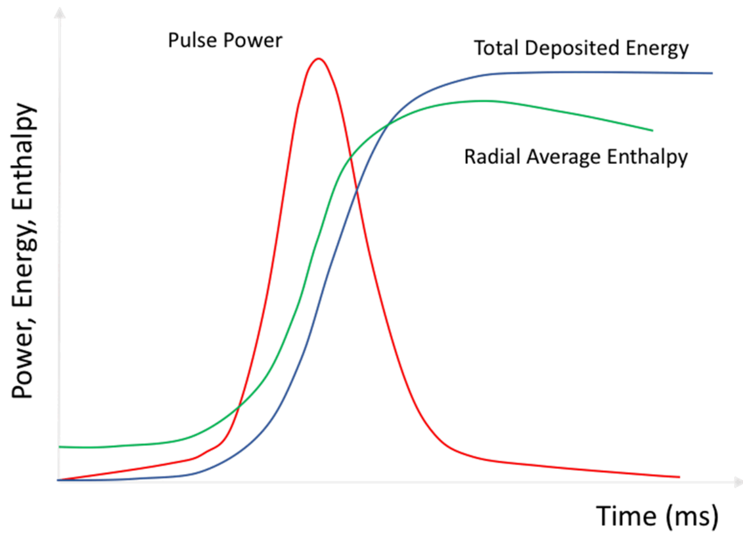

Figure 4. Typical power, energy, and enthalpy curves in an LWR design basis RIA.

The fissions in LWRs are primarily the result of thermal neutrons. As such, there is pronounced self-shielding effect where the fission density in the outer radial portion of the fuel rod is higher than in the center. During irradiation, the effect is exacerbated by the buildup of plutonium in the outer radial edge of the fuel. This burnup dependent phenomenon is referred to as radial power shift. The result of self-shielding and radial power shift in a design basis RIA transient originating from zero power is that the typical parabolic shape of the fuel's radial temperature profile is inverted. During the prompt phase of the transient, much higher temperatures are seen in the outer radial edge of the fuel as opposed to the fuel centerline. As heat begins to transfer to the coolant, the peak radial temperature location moves radially back towards the fuel centerline. Sufficiently severe transients can result in regions of localized pellet melting anywhere along the pellet radius, depending on the dynamic evolution of the transient and heat transfer conditions within the fuel rod.

As the fuel pellet heats up, rapid thermal expansion of the pellet takes place along this inverted radial temperature profile. When the pellet cladding gap has been partially or completely closed due to pellet swelling and cladding creep during normal operations, strong pellet cladding mechanical interaction (PCMI) can occur. The PCMI loading on the cladding is displacement controlled and limited to low overall strains, but with high strain rates. The loading is uniform across the circumference of the cladding, and the stress state is somewhere between a plane strain condition and an equibiaxial condition. If the friction between the pellet and cladding is high due to the formation of a pellet clad bond layer, then axial forces will be high, and an equibiaxial stress state will exist. If slip is allowed to occur between the pellet and cladding, the stress state will be close to that of plane strain. These stress states result in a high principal stress in the circumferential direction for a given effective stress and increase the probability of brittle fraction prior to any appreciable plastic deformation. PCMI-based cladding failure is generally characterized by a long axial crack in the cladding, graphically depicted in Figure 5.The probability of cladding failure due to PCMI loading is greatly increased by the environmental degradation of the cladding during its life in the reactor. Irradiation damage, cladding corrosion, hydrogen pickup, and closure of the pellet clad gap all play important roles, with hydrogen pickup and associated hydride embrittlement being the dominate environmental degradation mechanism for Zircaloy-based claddings. Intuitively, transients with a longer pulse width should have greater resistance to PCMI failures as the cladding ductility should increase in longer transients.

As heat generated in the RIA is transferred from the fuel to the cladding and ultimately the coolant, a dramatic increase in the cladding coolant heat flux occurs. It is possible that a boiling crisis can occur at these high heat fluxes, resulting in cladding temperature excursions. The heat flux at which the boiling crisis occurs depends greatly on the thermal hydraulic conditions of the reactor at the time of the transient and may be higher than the heat flux leading to a boiling crisis in slower power ramps. High cladding temperatures can result in significant annealing and/or softening of the cladding material, leading to high plastic deformation. The transient also generally results in the fragmentation of the fuel into small millimeter-sized fragments which also leads to the sudden release of fission gas previously held in the fuel matrix. Transient fission gas release when combined with the increase in the plenum gas temperature can result in a dramatic increase in the fuel rod internal pressure. The combination of high internal pressures and cladding deformation leads to swelling and rupture failures if the fuel rod's internal pressure surpasses the system pressure of the reactor. This is graphically illustrated in Figure 5. If the internal pressure of the fuel rod is below that of the reactor system, the opposite can occur, where the weak cladding is collapsed onto the fuel and between fuel pellets, resulting in areas of high stress concentrations upon cooling.

Even if the cladding does not fail due to PCMI or swelling and rupture type behaviors, if the transient critical heat flux is exceeded, the cladding is vulnerable to rapid environmental degradation due to exposure to high temperature steam. High temperature steam oxidation is known to significantly embrittle Zircaloy claddings. In fact, preventing or slowing this degradation is a main design feature of most ATF cladding concepts. Rewetting thermal stresses principally in the axial direction can result in cladding rupture along the circumference of the cladding. Even if the cladding does not rupture upon quench, the change in mechanical properties as a result of the temperature transient can be sufficient to embrittle the cladding such that post quench ductility (PQD) is no longer maintained. Past regulatory reviews have concluded that a coolable geometry cannot be assured without a minimum level of post quench ductility [12]. In especially severe transients, partial or complete melting of the fuel rod can also occur, leading to a loss of coolable geometry. These degradation modes are illustrated in Figure 5.

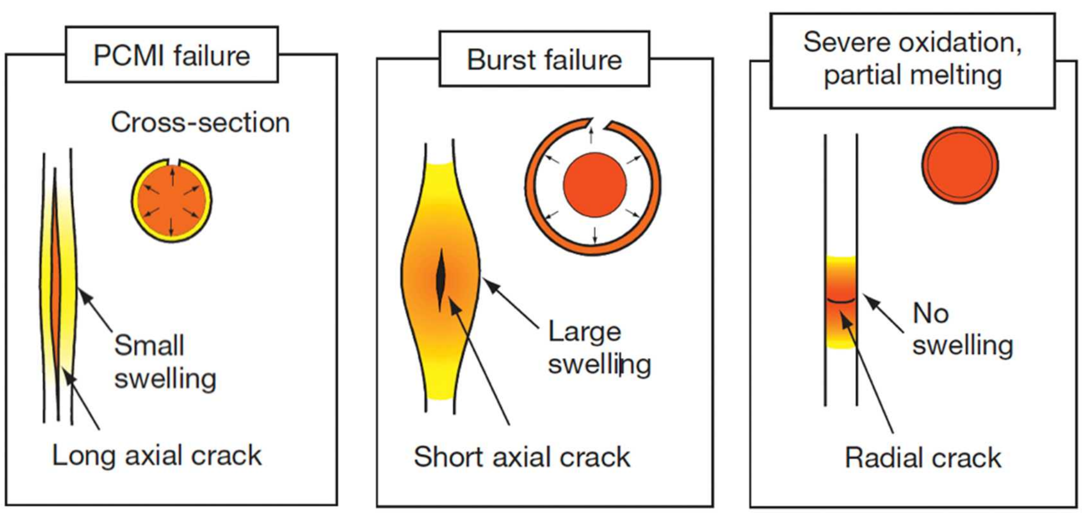

Figure 5. PCMI, Swelling and Rupture, and Severe Oxidation Failure Modes in RIA Transients [10].

An additional threat to the maintenance of coolable geometry in RIA transients is the generation of pressure pulses as the result of fragmented or molten fuel interaction with the coolant following a fuel rod failure. Fuel coolant interaction (FCI) can occur as the result of several of the transient fuel behaviors described above. Thermal to mechanical energy transfer can occur as the result of coolant vapor generation due to rapid heat transfer from the (molten) fuel particles and subsequent collapse of those vapor bubbles. These coolant pressure pulses have the potential to cause damage to neighboring fuel assemblies, core internals, and, potentially, the reactor pressure vessel boundary. The size of FCI pressure pulses is dependent on the thermal to mechanical energy conversion ratio which increases with higher fuel temperatures and finer particle sizes.

Nuclear fuel safety criteria for RIA events are generally reported in terms of a peak radial average enthalpy or a change in peak radial average enthalpy. Other relevant criteria are limits on cladding temperature, where it is generally assumed that fuel damage occurs if a boiling crisis occurs. For current designs, the initial enthalpy limit for fuel rods consisting of UO_2 pellets incased in a Zircaloy cladding were established in 1974 and published in the Atomic Energy Commission (AEC) Regulatory Guide 1.77. Initially, the coolability limit was set to 280 cal/g UO_2 radial average fuel enthalpy for both fresh and irradiated fuel [13]. This enthalpy limit was based on a review of Special Power Excursion Reactor Test (SPERT) and TREAT experimental data from the 1960's. Failure consequences were insignificant for *total energy depositions* below 300 cal/g-UO2 for a database of fresh and low burn-up fuels. It is critical to note the difference in terminology used by the Regulatory Guide 1.77, and the limit derived based on experimental observation – *radial average fuel enthalpy vs total energy deposition*. The difference is largely due to heat transfer from the rod during the event and heating from delayed fissions post-transient resulting in 10%-25% difference between radial average fuel enthalpy and total energy deposition (i.e., 300 cal/g-UO2 total energy deposition ≈ 240 cal/g-UO2 radial average fuel enthalpy in SPERT). Later studies during the RIA program performed at the Power Burst Facility (PBF) highlighted this discrepancy and extended the experimental database to include irradiated fuels up to mid-range burnups [14]. The reactor coolability criteria is now set at 230 calories per gram peak radial average enthalpy for LWR fuel consisting of Urania fuel pellets in a Zircaloy cladding [7]. (See Figure 6.)

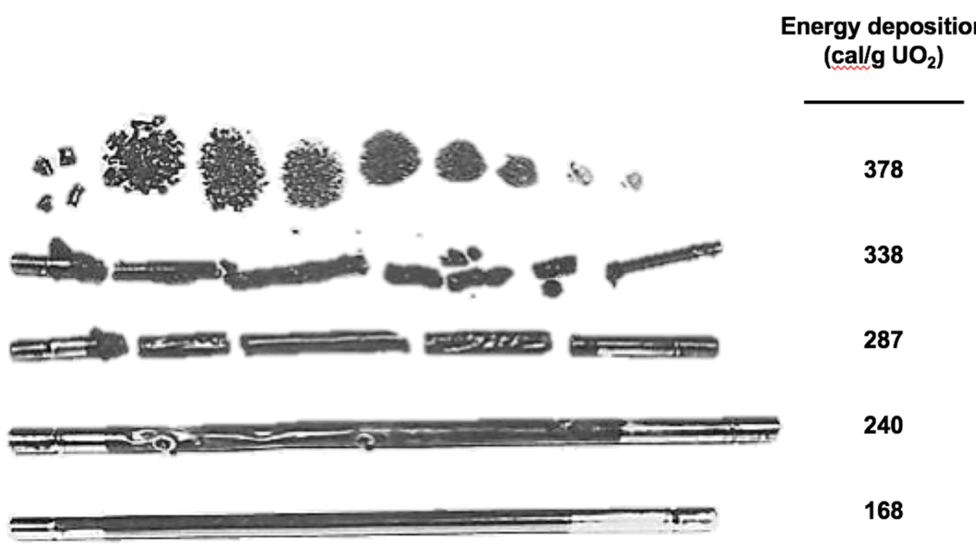

Figure 6. Pictures of failed fuel following RIA testing in SPERT.

Leading on from the PBF experimental program, testing of higher burnup fuels in the 1980's and 1990's performed in Russia, Japan, and France began to uncover PCMI as a new fuel failure mechanism for extended fuels with extended burnups. While fuel failure does not necessarily result in a significant challenge to coolable geometry, it can provide a pathway for the release of fission products which need to be quantified in plant safety analysis. Fuel rod failure is also a prerequisite to the release of solid or molten fuel particles to the reactor coolant, resulting in FCI pressure pulses. These pressure pulses are difficult to account for and analyze, and often plant safety analyses chose to avoid these issues by remaining below the fuel failure limit. The current RIA fuel failure limits are inclusive of low temperature failures (PCMI driven) and high temperature failures (swelling and rupture driven). The high temperature failure limit is either 170 cal/gram or 150 cal/gram depending on the state of the fuel rod internal pressure. The low temperature (PCMI) limit is based on the state of environmental degradation in the cladding as seen in Figure 7. When applying the PCMI criteria, only the change in enthalpy above the current fuel temperature at the beginning of the transient needs to be considered. The enthalpy limits are functions of cladding excess hydrogen, as that was found to be the dominating embrittlement mechanism for the Zircaloy-based claddings. Different curves are applied for stress-relieved Zircaloys versus recrystallized annealed Zircaloys based on the morphology of the hydrides that form and the observed difference in failure enthalpy. A 20 cal/gram shift is also applied when considering transients taking place at hot reactor conditions versus cold reactor conditions [15].

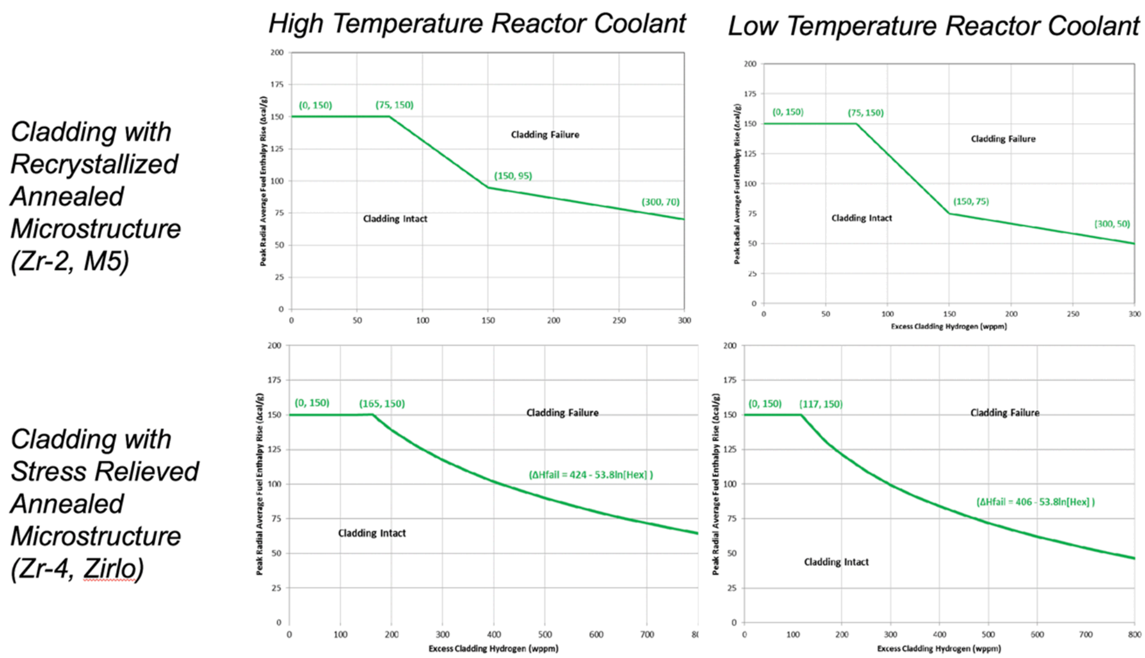

Figure 7. Current PCMI failure limits for Zircaloy claddings [15].

2.3 Transient Behavior and Fuel Safety Criteria for Loss of Coolant Accidents

Loss of coolant accidents (LOCAs) are a family of transients where the reactor system experiences a sudden and unexpected loss of cooling due to a rupture somewhere in the primary circuit of the nuclear steam supply system (NSSS). LOCAs are generally split into large break and small break LOCAs. A large break is generally considered any rupture larger than 1 square foot. A small break is one that is less than a foot squared but greater than circular breaks 3/8-inch in diameter. Breaks smaller than this can generally be made up for by the primary reactor coolant charging system. In PWRs, the design basis LB-LOCA is an assumed double-ended break in one of the cold legs between the primary reactor coolant pump and the outer annulus of the reactor vessel. For a BWR, the design basis LB-LOCA is a double-ended break in the suction side of one of the recirculation loops. The main phases of the LOCA event are: (1) Blowdown, (2) Refill, and (3) Reflood. Figure 8 shows a qualitative plot of fuel cladding temperature and reactor power during an accident progression.

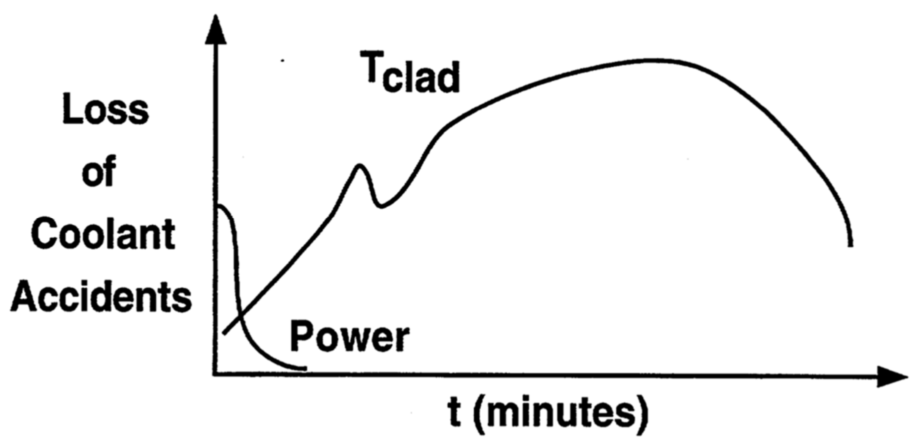

Figure 8. Generalized nuclear fuel power / temperature plots during a LB-LOCA [12].

During the blowdown phase, most of the primary coolant is expelled through the break. Voiding in the core causes a shutdown of the reactor. The reactor pressure rapidly drops from an operating value of ~15.5 MPa (PWR) to near atmospheric pressure of 0.1MPa. High pressure safety injection begins; however, most of this coolant is swept around the downcomer and lost out the break. Numerous flow reversals in the core can occur as the core is initially cooled by a resulting two-phase mixture. As the two-phase coolant flow in the reactor drops and the stored energy of fuel redistributes from the fuel pellet to the cladding, the core heats up very rapidly. Following the blowdown phase of the LOCA, the ECCS begins to refill the core. Initially, the bottom plenum of the core must be refilled so during this time the reactor core continues to heat up in a near adiabatic fashion due to the decay heat being generated in the fuel. The heatup of this phase is generally around 10s of °C per second. Eventually the ECCS completes the filling of the reactor's lower plenum, and coolant begins to move upward and reflood the core. The rate of coolant rise through the core can be very slow, as low as 1 inch per second for some reactors. Initially, the peak cladding temperature continues to increase until a sufficient part of the core is covered. As the lower elevations of the core quench, a two-phase mixture of steam and entrained water droplets provide some enhanced heat transfer to the upper parts of the core. Quench occurs when the liquid water front moves past the fuel rods at a given axial location.

The study of LWR fuel performance in LOCAs has a long history going back to the beginnings of LWR technology [16]. The understanding of fuel performance during LOCAs, the associated safety criteria, and the testing approach have all evolved and matured through time. The principal transient fuel behaviors in LOCAs are (1) swelling and rupture, (2) fuel fragmentation relocation and dispersal (FFRD), and (3) loss of post-quench ductility due to high temperature steam oxidation. These phenomena are similar to some of those discussed in the RIA section above; however, given the longer time frames involved in a LOCA transient, their progression can be somewhat different. A general overview of how these phenomena play out in a LOCA transient is given below. Recommended general overviews of LOCAs from a fuel performance perspective include:

- "Nuclear Fuel Behavior Under Loss of Coolant Accident (LOCA) Conditions" [17]
- "Behavior of LWR Fuel During Loss-of-Coolant Accidents" [18]
- "Report of Fuel Fragmentation Relocation and Dispersal" [19]
- "Phenomenon Identification and Ranking Tables for Loss of Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel" [12].

The high-pressure differentials and high temperatures that occur in a LOCA make the cladding highly vulnerable to swelling and rupture behavior. In isotropic materials deforming at isothermal temperature, the deformation is unstable. As the tube expands radially, the thickness will decrease to conserve volume. The combination of larger diameter and lower thickness has a positive feedback effect and will lead to early rupture. In these situations, it is expected that cladding swelling and rupture locations will be distributed randomly over the axial length of the cladding. However, under cooling conditions a thermal stabilizing process causes a negative feedback through increased cooling of the larger surface area of the strained cladding. As creep strength is a strong function of temperature, the cooler cladding will have increased resistance to further deformation. In addition, nuclear fuel rods are heated internally; thus, as the cladding deforms radially outward from the fuel, the increased insulation from a larger gas gap will lead to a further cooling of the cladding in the ballooned region. These negative temperature effects can result in larger axially aligned balloons occurring at the axial location of highest temperature, generally between 2/3 and 3/4 of the way up the core. Large axially aligned balloons are a challenge to core geometry.

Zircaloy fuel cladding is both cold work and partially annealed, resulting in anisotropic behavior, especially in its unirradiated state. The crystal structure is Hexagonal Close Packed (HCP) which is referred to as the alpha phase. The <0001> poles are aligned ~30° off of the radial direction (Figure 9) resulting in a strong anisotropy that resists wall thinning. When the cladding begins to deform radially, it does so by axial material flow. In cases where the deformation has an azimuthal variation, this anisotropy will cause one side of the cladding to shorten more than the other, causing a bowing of the hot side against the pellets. This is known as the "hot-side straight" effect, and the bowing caused by this non-uniform expansion can be a further threat to coolant channel blockage.

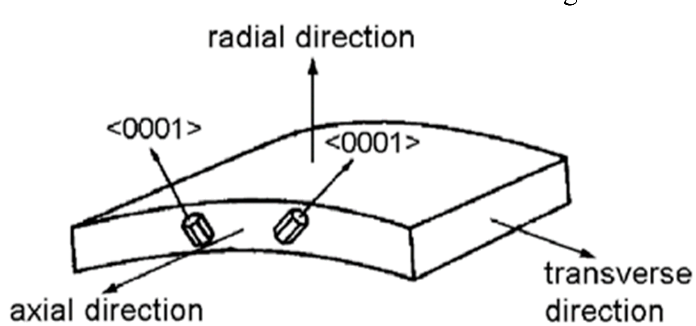

Figure 9. Schematic of grain orientation in Zircaloy cladding [20].

Zircaloy begins to undergo a phase change at ~800 °C from the HCP alpha structure to a Body Centered Cubic (BCP) beta phase structure. Between ~800 °C and ~980 °C a two-phase region exists, and above ~980 °C, a pure beta phase exists. The mixed phase region and beta phase region have significantly lower creep strength than the alpha phase, and, unlike the alpha phase, the beta phase behaves isotopically. As such, strain rates in parts of the cladding exhibiting higher concentrations of beta phase grains are likely to be much higher. Zircaloy used as nuclear fuel cladding is an alloy of zirconium with minor alloying additions of tin, niobium, iron, chrome, nickel, and oxygen. The extent of the alloying mixture will affect the alpha/beta transition which will in turn affect the deformation behavior. Some elements like niobium, chrome, and iron are beta stabilizers, so they will have the effect of lowering transition temperatures. Other elements like Tin and Oxygen are alpha stabilizers, so their effect will be the reverse. Early rupture at low overall strain will occur in cases where local ductility is exhausted by non-uniform straining. This occurs when axial or azimuthal temperature variations lead to non-uniform straining. Azimuthal temperature variations can occur in the nuclear fuel due to varying temperatures of neighboring rods, the pellet stack not remaining co-axial during the balloon, axial fuel fragment relocation into the ballooned region, or due to the cladding's anisotropic behavior.

While not seen in lower burnup fuels, recent experiments involving high burnup fuel have shown that the fuel is vulnerable to fine fragmentation, sometimes referred to as pulverization, in the swollen and ruptured region of a fuel rod. This phenomena in LOCAs is referred to as fuel fragmentation relocation and dispersal (FFRD). FFRD can present significant challenges in terms of quantifying a release fraction in the accident analysis as well as providing assurances of coolable geometry and reactivity control, as a significant portion of the core may end up in a debris bed like configuration. The cause of FFRD is unknown and investigations at many different research institutes are ongoing. Some leading theories are that it is caused by either over pressurization of fission gas bubbles that are present in the dark zone of the fuel pellet immediately adjacent to the high burnup structure that forms on the outer radial surface, or that thermal stresses as the result of the power history and furnace heating conditions present in most experiments are initiating pulverization process. There are some data to support that the burnup threshold for FFRD is somewhere between 62 and 75 GWD/MT. Other thresholds for FFRD include a maximum amount of hydrostatic pressure above which fragmentation does not occur, a minimum fuel temperature below which fragmentation does not occur, and a minimum last cycle linear power below which fragmentation does not occur. A summary of these thresholds and how they impact the LOCA safety evaluation including effects on coolable geometry, fission product release, and effect on cladding temperature is shown graphically in Figure 10.

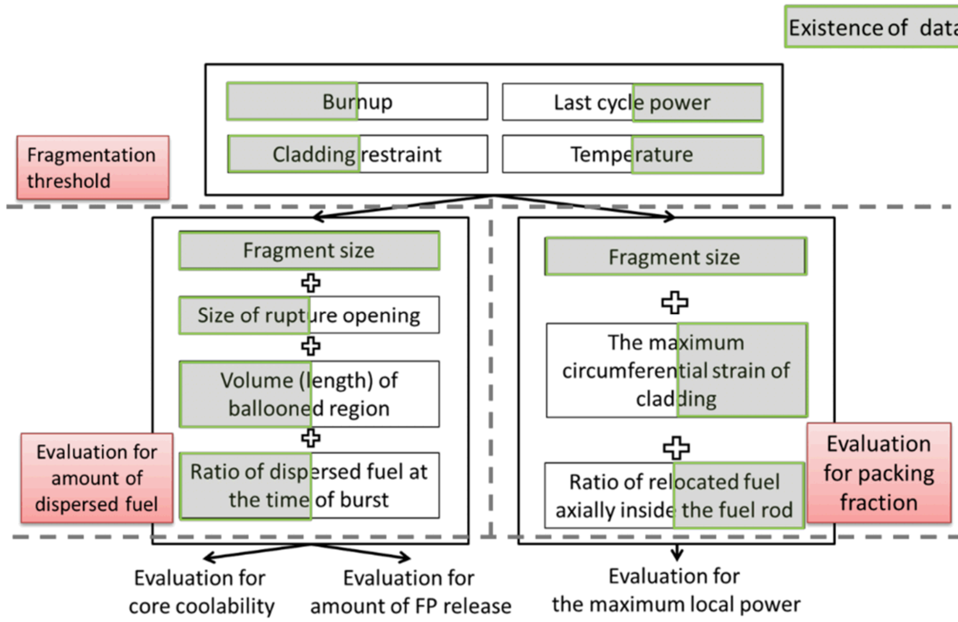

Figure 10. Potential FFRD thresholds and their effect on LOCA evaluation figures of Merit [19].

During the reflood stage of the LOCA, a general period of isothermal high temperature steam oxidation can occur in the top parts of the core. The oxygen attack on the Zircaloy cladding at high temperatures can significantly embrittle the cladding and can be a significant challenge to the post quench ductility of the cladding which is required to ensure a coolable geometry. The oxidation in high temperature steam is significantly different than that experienced during normal operations. In normal operating conditions, the Zircaloy cladding initially experiences oxidation according to a cubic growth law until a sufficiently thick oxide layer has built up on the cladding, at which point it transitions to a linear growth. The transition thickness is generally in the range of 2 µm. The oxide that grows under normal operating conditions is initially a thick tetragonal oxide. However, it quickly transitions a monoclinic oxide with columnar grains. The monoclinic oxide is generally seen as less protective then the tetragonal oxide, although this may be due to cracking of the oxide due to compressive stresses. During corrosion, 15 – 20% of the hydrogen that is liberated in the oxidation reaction is also absorbed into the cladding. This is known as the hydrogen pickup fraction. Hydrogen solubility in Zircaloy is relatively low (80 wppm at 300°C) and a strong function of temperature; however, it has a high mobility, so the dissolved hydrogen often precipitates as brittle circumferentially oriented hydrides in cooler regions of the cladding at the outer surface (known as the hydride rim) and at axial locations between pellet boundaries.

During high temperature steam oxidation, a dense tetragonal oxide grows rapidly between the oxide formed during normal operations and the base metal according to a parabolic rate law. The hydrides dissolve at high temperature and migrate to the beta phase where they stay in solid solution. During the high temperature oxidation reaction, significant amounts of additional hydrogen is also liberated. However, the high temperature oxide is generally protective against addition hydrogen pickup resulting in most of the liberated hydrogen building up in the reactor core. If a significantly high concentration of hydrogen builds up in the core there is a risk of detonation which is a significant challenge to plant safety. During high temperature oxidation, oxygen also diffuses into the base metal and, due to its low solid solubility in beta zirconium, it transforms the Zircaloy back to the alpha phase but with a very high oxygen concentration. Figure 11 graphically shows the differences in oxidized Zircaloy cladding microstructure between normal operations and high temperature steam oxidation. At high temperatures, the cladding also oxidizes from the inside, either due to the contact with the hyper stoichiometric UO_2 on the fuel pellet rim or due to steam ingress near the ruptured region of the cladding.

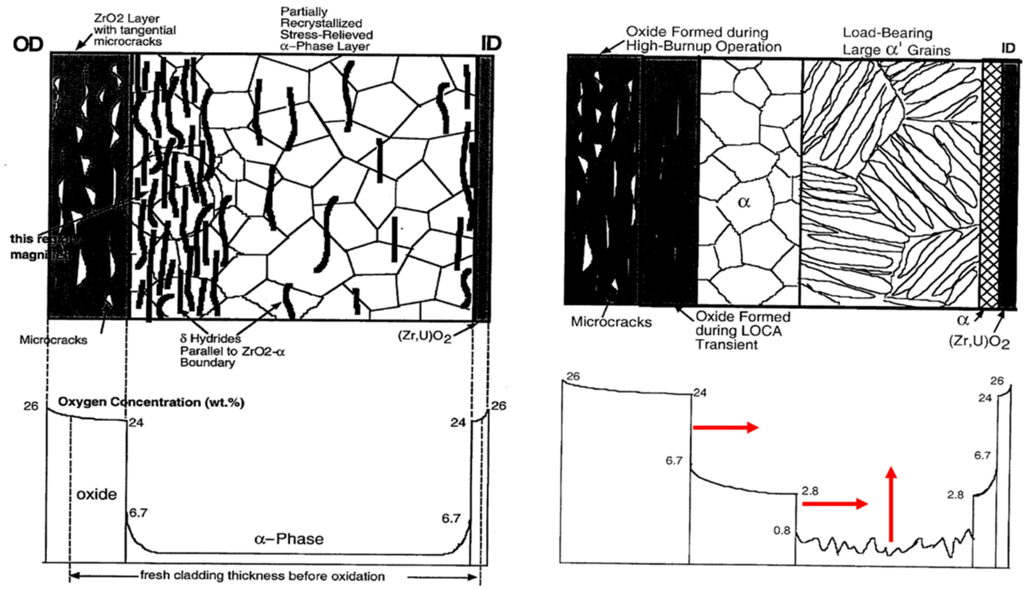

Figure 11. Zircaloy cladding oxidation in normal operations (left) and in high temperature steam (right) [12].

It is generally accepted that a minimum thickness of the beta phase with less than a maximum oxygen concentration is required for the cladding to maintain its strength and ductility following the LOCA. The oxygen attack on the cladding can be enhanced by a number of other phenomena that occur prior to and during the LOCA. It has been shown that the presence of hydrogen in the Zircaloy cladding can significantly enhance oxidation induced embrittlement [21]. Hydrogen is highly soluble in beta zirconium and has the effect of raising the solubility of oxygen in the beta phase as well, decreasing the time and temperature required to saturate the beta phase with oxygen above what is known to embrittle this phase. The high hydrogen partial pressures inside of a burst fuel rod can result in a high hydrogen pickup on the inside of the cladding. This phenomenon is known as secondary hydrogen pickup. Secondary hydrogen pickup can dramatically reduce cladding ductility around the rupture opening in the cladding. Finally, as the high temperature oxide grows, it too can form cracks as the result of compressive stresses which leads to an acceleration in the oxide growth kinetics from a parabolic rate law to a linear rate law. This phenomenon is known as breakaway oxidation. Breakaway oxidation thresholds are highly dependent on the zirconium alloy and the surface characteristics of the cladding material. Cladding is most vulnerable to breakaway oxidation in specific temperature bands generally found around 900°C and 1100°C. In breakaway oxidation, not only is the oxidation rate increased but the oxide layer loses its protective features, resulting in additional hydrogen pickup. The cladding cools rapidly once it is covered in liquid water in the reflood. This quench from high temperature can introduce large thermal mechanical loads on the cladding and can affect the final microstructure of the Zircaloy which can affect its post-quench mechanical properties.

Safety criteria for LOCAs are presently found in 10 CFR 50.46 and are formed to prevent oxidation induced embrittlement, limit overall hydrogen generation in the core, and ensure a coolable geometry. FFRD behavior is not addressed in the current criteria, as this behavior is not observed below the current burnup limit of 62 GWD/MTU. To ensure coolable geometry, each vendor's fuel performance code must include a validated swelling and rupture model to account of changes in core geometry during the LOCA.

An example of an approved model is found in [22]. A maximum peak cladding temperature of 2200°F (1204°C) and an effective cladding reacted (ECR) limit of 17% are set to ensure post-quench ductility of the Zircaloy cladding. ECR is an analytical value that is determined by assuming that all of the oxygen consumed in the oxidation reaction forms stoichiometric Zirconium Oxide, the amount of zirconium that would be consumed in this assumed reaction as a percentage of the zirconium present is the ECR. These limits were set based on data from out of pile furnace testing at ANL [20] and in pile data from the Power Burst Facility (PBF) [23] and shown below in Figure 12. It is assumed that the 17% ECR ensures a minimum thickness of the prior beta layer, and the 1204°C limit ensures that the oxygen concentration in this layer is sufficiently low to preserve its mechanical strength.

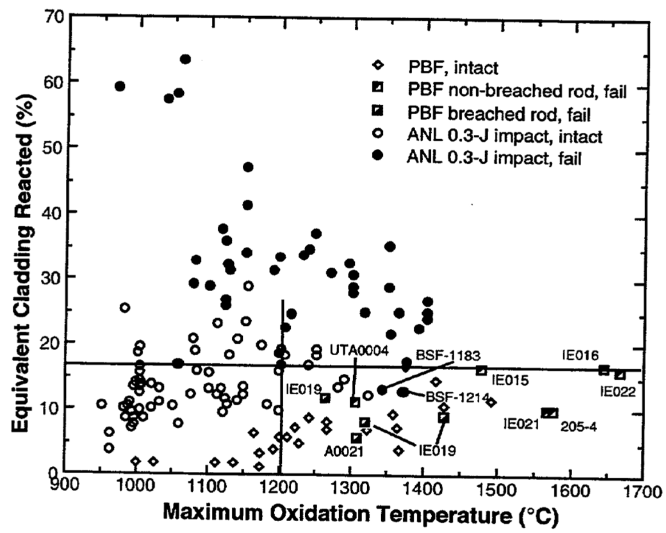

Figure 12. Cladding ductility data from ANL furnace tests and in-pile PBF tests [12].

More recent data from NRC's LOCA research program at ANL have shown that these limits may not always be conservative from high burnup cladding materials with high hydrogen concentrations [21]. This has prompted a controversial rule change proposed by NRC called 50.46(c) which requires fuel vendors to establish analytical limits for the performance of cladding materials based on results from standard post-quench ductility tests [24] and combined reactor system and fuel performance analytical tools. Examples of these limits as a function of hydrogen content for the modern Zircaloy cladding types along with the data used to develop them are given in [25] and shown below in Figure 13.

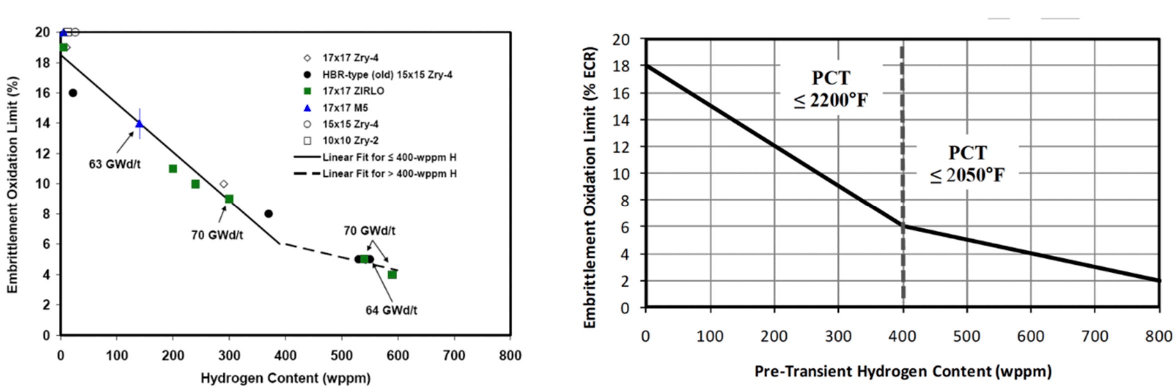

Figure 13. Analytical limits for Zircaloy cladding in LOCAs with supporting data.

2.4 Transient Behavior and Fuel Safety Criteria for Generalized Power Cooling Mismatch Scenarios

Both LOCA and RIA transients represent specific categories of events that involve a mismatch between the reactor power and the ability to provide adequate cooling. RIAs generally represent one end of these spectrum of events where there is a very brief sharp increase in reactor power, however, the reactor coolant systems are assumed to operate normally. LOCAs represent the other end of the spectrum where the reactivity and, thus, reactor power is controlled, but there is an inability to provide cooling to the core. Design basis RIAs are generally very brief in nature lasting from milliseconds to seconds. Design basis LOCAs can be much longer lasting from minutes to hours.

There are many commonalities between the transient fuel behaviors observed in LOCAs and RIAs generally, as the result of the boiling crisis that occurs in both events. Behaviors like cladding swelling and rupture and high temperature steam oxidation occur in both events. Other similarities exist between behaviors like FCI in an RIA and FFRD in a LOCA that both involve the expulsion of fissile material from the fuel rod and into the core in localized regions of fuel failure. Although there are similarities in the observed phenomenon, the fuel performance limits in each scenario are likely to be different given that the nature of the transients are different. As an example, the temperatures seen in LOCA are generally lower and the temperature swings more gradual, and, as such, the thermal stresses are generally lower and fuel and cladding materials are unlikely to reach their melting point. In contrast, the duration of RIA events is generally too brief for phenomenon like breakaway oxidation or secondary hydrogen pickup to impact the oxidation kinetics.

Generalized Power Cooling Mismatch (PCM) transients are a broader group of transients that are meant to cover the space between RIA and LOCA transients as illustrated in Figure 14. A notable characteristic of a generalized PCM event is that it does involve a severe enough mismatch in power and cooling that a boiling crisis results. Transients which are not severe enough to cause a boiling crisis are referred to as power ramps or overpower events and often have a unique set of often more subtle transient fuel behaviors. Depending on the frequency and severity of the PCM transient, it may be classified as a DBA or an AOO. It is often practice when analyzing an AOO event to assume that a fuel rod has failed if it experiences a departure from nucleate boiling (DNB). While this is acknowledged as conservative, there is generally a lack of data to support an alternative safety criterion for AOO events that would ensure that fuel rod damage was sufficiently minimal as to ensure its continued operational use.

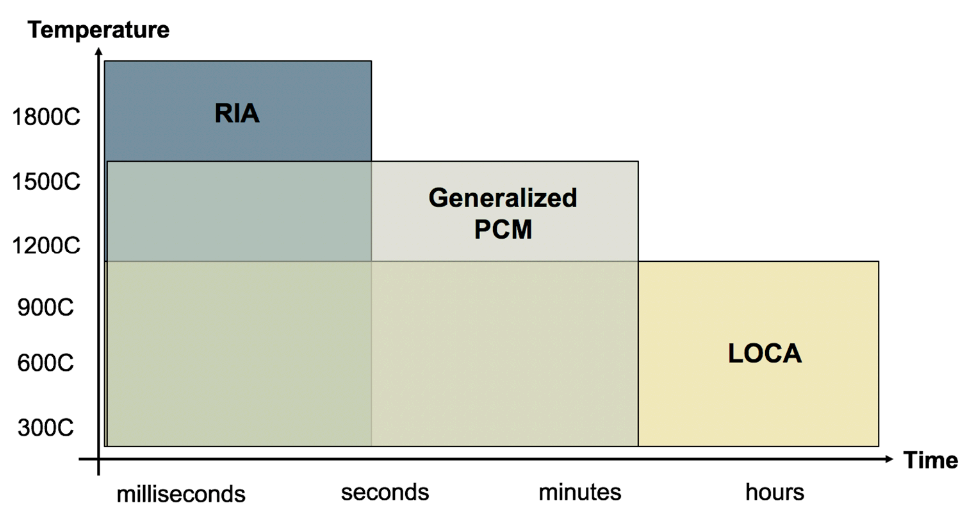

Figure 14. Time temperature domains of various LWR transients that involve a boiling crisis.

Generalized PCM scenarios generally receive less attention in textbooks, regulatory discussions, and experimental priority than RIAs and LOCAs. Nevertheless, important transient fuel behaviors have been identified through testing in more generalized AOO events. The PCM and IE experiment series at the PBF reactor focused on these kinds of scenarios and made important contributions to our understanding of oxide fuel pellet restructuring, fuel centerline melt including its effect on cladding strain, cladding collapse, and high temperature steam oxidation of Zircaloy cladding [23] [26]. Studying general PCM events helps bridge some of the gaps in understanding of transient fuel behavior that would exist if only RIA and LOCA events were considered.

3 TESTING MATERIALS AND METHODS

Developing an understanding of transient fuel behavior requires testing with materials that are representative of those in commercial reactors and utilizing testing methods that closely replicate the anticipated transient events. Several different ATF designs are being pursued by the nuclear fuel vendors in the U.S., and some of these concepts are under irradiation in LTA programs at commercial reactors. The fuel safety program is working closely with each vendor to obtain materials that are representative for the various transient tests proposed. Additionally, efforts are well underway to develop testing methods and infrastructure to enable testing of these materials in representative environments.

3.1 Accident Tolerant Fuel Materials

The U.S. nuclear fuel vendors have proposed ATF materials which include both fuel and cladding materials. In addition to standard UO_2 pellets, ATF pellet materials include doped UO_2, and U_3Si_2. In addition to standard Zircaloy cladding, ATF cladding materials include Iron Chrome Aluminum alloys (FeCrAl), Silicon Carbide fiber composite matrix structures (SiC-SiC), and Zircaloy claddings with thin coatings (usually chrome metal). An in-depth discussion about the various potential benefits and drawbacks of the ATF concepts is beyond the scope of this research plan. The Nuclear Energy Agency (NEA) has published a State-of-the-Art Report on Accident Tolerant Fuel [27] which includes more details on these concepts as well as others under development. In addition, several overview papers have been recently published that address specifically the ATF cladding materials [28][29][30]. A brief overview of these materials is included below. The emphasis is on the concepts perceived transient fuel behavior.

3.1.1 Doped (Large Grain) UO_2 Fuel Pellets

Doping UO_2 with ppm levels of Chromia (Cr_2O_3) and/or Alumina (Al_2O_3) results in higher grain sizes during sintering which results in a softer pellet which can decease the mechanical force transferred from the fuel to the cladding during pellet clad interaction (PCI). Another potential advantage is that the larger grain sizes may increase the retention of fission gases in the fuel matrix which may lead to either increased or decreased transient fission gas release depending on the mechanism of fission gas bubble retention [31]. Doped fuel pellets are being developed by both Framatome and Westinghouse.

3.1.2 Uranium Silicide (U_3Si_2) Fuel Pellets

Uranium Silicide (U_3Si_2) has dramatically different thermal, physical and mechanical properties from UO_2 [32]. The melting point is much lower (~1700 °C) as is the high temperature heat capacity (217 J/kgK at 1000°K versus 312 J/kgK for UO_2). The thermal conductivity, however, is much higher (23 W/mK at 1000°K versus 5 W/mK for UO_2). Fission gas retention is largely unknown, although one could theorize that lower temperatures lead to higher fission gas retention. Early irradiations in the ATR seem to provide some validation that the lower temperatures experienced by U_3Si_2 during irradiation do lead to lower fission gas release. Variation in mechanical properties with irradiation are also unknown. The lower heat capacity and lower melting point indicate that the likelihood of fuel melting is significantly increased with this fuel type. Therefore, this concept may be vulnerable to transient situations where the heating rate is higher than heat rejection times for the nuclear reactor as in the case of design basis RIAs. The increased thermal conductivity and uranium loading of U_3Si_2 fuel have many potential benefits in normal operations and in transients where sufficient heat removal is provided. Uranium Silicide fuel is primarily being developed in the U.S. by Westinghouse.

3.1.3 Coated Zircaloy Claddings

Coating Zircaloy claddings is done primarily to reduce the oxidation reaction of the Zircaloy with water and increase the hardness to improve the claddings fretting and wear resistance. The effect is a decrease in corrosion related embrittlement mechanisms such as hydride embrittlement as well as a decrease in fretting failures. The coatings are also shown to be protective in reducing high temperature steam oxidation of the cladding [33]. These ATF materials could thus provide a significant enhancement in material performance following transients that exceed the reactor's critical heat flux. The decrease in hydride related embrittlement of these claddings could also significantly minimize the PCMI failure probability. The thin coating does not significantly alter the bulk material properties of the cladding, and, thus, the concept is still vulnerable to swelling and rupture type failures. Some empirical evidence has recently been developed which indicates that the balloon sizes and rupture openings of coated claddings may be smaller than uncoated claddings [34]. The coatings are generally not applied on the inside of the cladding and, thus, would not protect the cladding from oxidation inducted embrittlement from the high temperature steam reacting on the inside of the cladding following rupture. General Electric, Westinghouse, and Framatome are all developing variants of coated Zircaloy cladding.

3.1.4 Iron Chrome Aluminum (FeCrAl) Claddings

FeCrAl claddings provide an increased level of corrosion resistance during normal operations and in high temperature steam environments. The alloy is also more resistant to high temperature creep than Zircaloy [35]. The alloy may have significant benefits in RIA performance in reducing the likelihood of balloon and burst type failures as well as high temperature oxidation induced embrittlement. The alloy does have a lower melting point than Zircaloy (around 1450 °C) which may introduce a new failure mechanism for this alloy in transients that result in high cladding temperatures. In addition, FeCrAl alloys are vulnerable to embrittlement from irradiation assisted segregation of brittle chrome rich alpha prime phases in the metal matrix. This irradiation induced embrittlement mechanism may increase the claddings susceptibility to PCMI like failures at higher burnups [36]. General Electric is the primary U.S. developer of FeCrAl claddings.

3.1.5 Silicon Carbide Composite (SiC-SiC) Claddings

SiC-SiC Composite claddings are remarkable both in their resistance to high temperature steam oxidation and in their mechanical strength at high temperature. Thus, the cladding is expected to perform much better than current materials in high temperature transients. The higher thickness and lower thermal conductivity of the cladding, however, could lead to higher temperatures in the fuel pellet which may result in increases in fission gas release, decreases in the fragmentation threshold, and even fuel melt. The concept is perhaps most vulnerable to displacement controlled PCMI loadings. While SiC is a brittle ceramic, the fiber-based structure of the composite is designed to provide some macroscale ductility in the structure. Out of pile mechanical tests show that the cladding does fail during displacement loadings mimicking PCMI; however, the general "rod-like" geometry is maintained indicating that core coolability criteria could still be met [36]. Westinghouse and Framatome are both developing variants of Silicon Carbide Claddings.

3.2 Previously Irradiated Materials

In order to understand the effect of burnup on transient fuel performance, it is necessary to conduct transient testing on previously irradiated materials. The most ideal source of previously irradiated material is from LTA programs in a commercial reactor. These materials have the most prototypic irradiation history. During irradiation in a LWR, the fuel and cladding are subject to a variety of nuclear, mechanical, and environmental degradation modes that may impair the performance of the fuel rod in a transient compared to that of a fresh rod.

Upon receiving a commercial LWR fuel rod for testing at TREAT, initial characterization of the rod will first be completed. Some of these characterization methods are destructive in nature. Therefore, it is often desirable to have two rods of similar power histories sent for testing, a test rod and a sister rod, which can be destructively examined. Details of the specific exams requested will be prepared in a separate Pre-Transient Material Characterization Report. Some important characteristics of LWR fuel rods that are important to impacting their transient performance are included in a bulleted list below for reference:

- Cladding oxide 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, and amount of fission gas in plenum
- State of pellet cladding gap
- Burnup and Isotopic inventory of fuel

The commercial LTA rods will be received and characterized at INL's Hot Fuels Examination Facility (HFEF). Following characterization of the test rod and sister rod, the test rod will need to be refabricated prior to the transient test. The rod will be cut into lengths appropriate for the transient test train. Following cutting, fuel will need to be removed from the ends of the rods to provide ample room for endcaps and instrumentation. The oxide layer will also need to be removed from the ends of the cut rod. Following cutting and cleaning, any internal instrumentation or spacer material will need to be inserted into the rod, and endcaps will be welded on. Most refabricated rods will need to be welded under pressure; several techniques for pressurizing the rod are currently under investigation. Following end cap welding, any external instrumentation will be attached to the rod, and it will be assembled in the transient test train. Depending on the complexity of the refabrication operation, the steps following cutting and cleaning may be done in HFEF or may be done in a dedicated facility with a modular hot cell which is currently under consideration for installation at INL and shown conceptually in Figure 15.

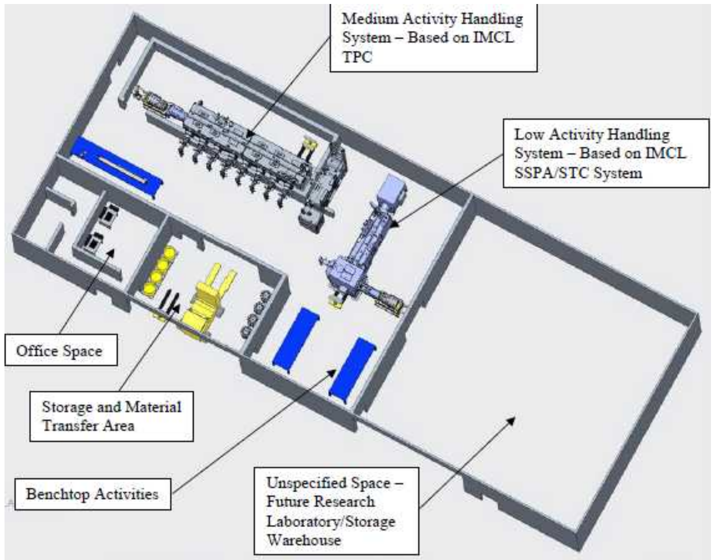

Figure 15. Test rod refabrication laboratory conceptual rendering.

3.3 Transient Irradiation Testing at TREAT

The TREAT facility was constructed in the late 1950's and provided thousands of transient irradiations before being placed in standby in 1994. The reactor resumed operations in 2017 in order to reclaim its crucial role in nuclear-heated safety research. TREAT is an air-cooled reactor driven by a core of graphite blocks having a small concentration of dispersed uranium dioxide. Columns of these graphite-fuel blocks are hermetically encapsulated in Zircaloy sheet metal canisters. Aluminum-sheathed unfueled graphite blocks are attached to the top and bottom of each fuel column; forming a discrete fuel assembly with 1.2m of active core length. Along with control rod, experiment, and graphite reflector assemblies, these fuel assemblies are placed on a 19 × 19 grid-plate with 361 available positions; creating a core that can be adjusted to suit particular nuclear parameters or experimental objectives. A few fuel assemblies are typically removed from the central core positions to create a cavity for experiments. Experiment assemblies are typically removed from or placed into the core through a slot in the reactor's upper rotating shield plug, handled outside the reactor using shielded casks, and stowed below grade in storage holes when not in use. Four horizontal access holes can be opened through the vertical concrete shield walls and permanent graphite reflector surrounding the above-grade core to provide various capabilities. Two horizonal access holes are currently in use, one for a neutron radiography beam, and the other for a fuel motion monitoring system (commonly referred to as the hodoscope) capable of collecting spatially resolved data from fast neutrons born in experiment specimens. A ¾ view of the TREAT reactor is illustrated in Figure 16.

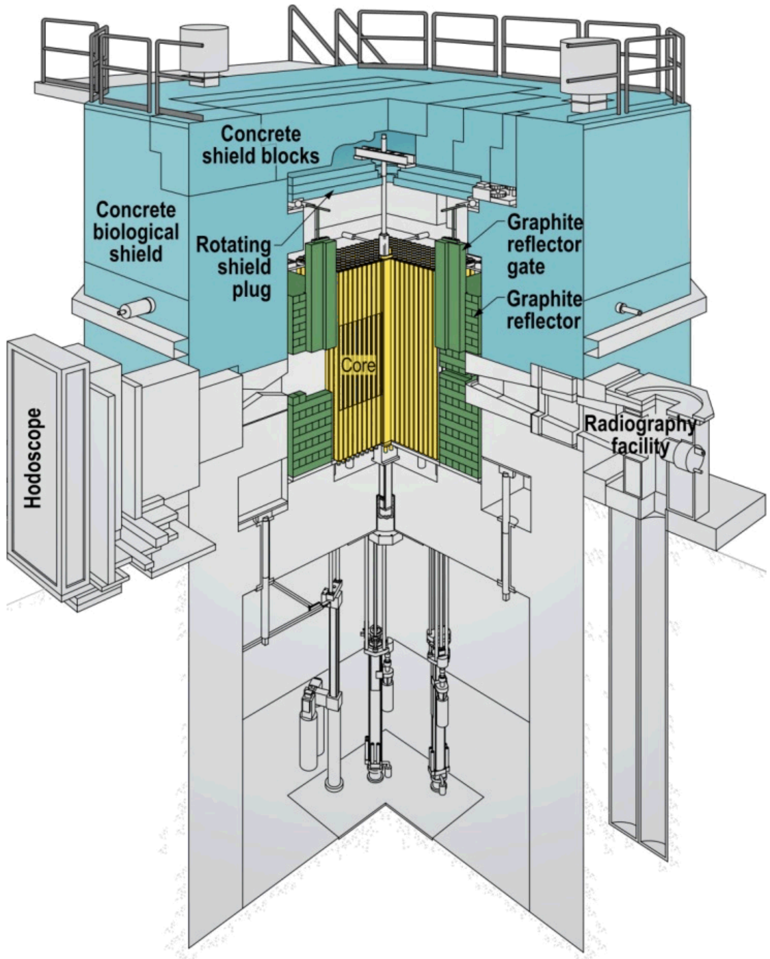

Figure 16. Cut away view of the TREAT reactor.

Pulse type transients are initiated in TREAT by bringing the reactor to a low steady state power of 50 watts and then rapidly removing transient control rods, resulting in a step insertion of excess reactivity. TREAT pulses initially have a nominally Gaussian shape followed by a decaying exponential tail. Larger step reactivity insertions result in transient pulses that have higher peak powers, higher overall energy releases, and shorter pulse widths. The fraction of the total energy that is released in the Gaussian part of the transient also increases with higher step insertions. A parameter called the Gaussian Energy is defined from the peak power of the transient and the FWHM of the transient and is constructed assuming a perfectly Gaussian power pulse. The Gaussian Fraction or fraction of energy released in the Gaussian part of the transient is the ratio of the Gaussian Energy to the total 60 second reactor energy release. The energy released from a TREAT transient 1 FWHM past the peak power is generally very close to the Gaussian Energy value [37].

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

TREAT has the ability to re-insert the transient control rods and shorten the natural transient. The clipping system has the ability to reduce both the pulse width (FWHM) and total energy deposited. As an example, for a 4.5% Δk/k reactivity insertion, clipping capabilities could decrease the maximum energy released from the reactor from ~2800 MJ to ~630 MJ and shorten the pulse width from 103ms to 95ms [38]. Because of TREAT's low starting power levels, initial changes in power are small even for very short reactor periods. As a result, the rapidly ejected transient rods are fully withdrawn from the core and resting at zero velocity before reactor power reaches appreciable levels, and clipping is triggered by a predetermined time delay in the transient programming. This approach gives the ability to tailor transients to a desired duration and energy release. However, the response is limited by the speed of the rod drive system (~355 cm/s). For TREAT's ~1m reactor length, it takes ~280ms for the control rods to fully insert themselves. Core energy release is highly dependent on the accuracy and repeatability of clipping maneuvers; especially when clipping occurs during the rise-to-power segment of an otherwise natural pulse. The current capabilities of the TREAT reactor make getting truly prototypic pulse widths for RIAs in PWRs at zero power conditions (approximately 30 ms) just out of reach. However, pulse widths of 89 ms have recently been demonstrated, and future plant modifications involving a He-3 clipping system are under investigations and are predicted to be able to achieve pulse widths as low as 45 ms. Figure 17 compares natural and clipped TREAT transients with those of LWRs and other transient reactors in terms of their pulse width and total energy release.

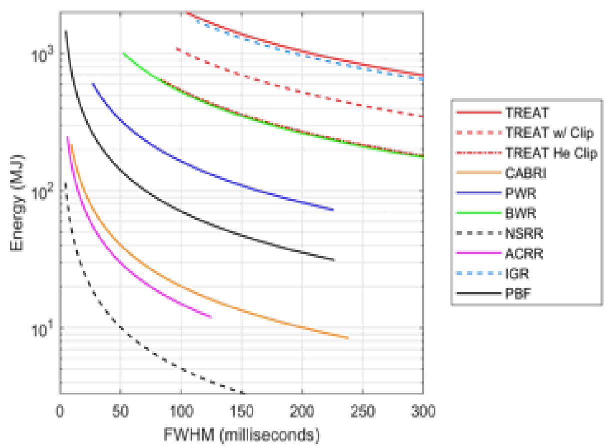

Figure 17. TREAT pulse characteristics.

TREAT also has the ability to operate in a "Shaped Transient Mode" where control rods can be manipulated in real time to create any arbitrary reactor power shape. Figure 18 shows the reactor power and control rod motion required to simulate a LOCA transient. When creating a shaped transient, TREAT has the ability to release approximately 2500 MJ of energy. TREAT also has the capability to couple the reactor control system to experiment lead outs such that reactor power can be manipulated based on the output from experiment instrumentation. As an example, reactor power could be manipulated to achieve a desired 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(energy) to specimen power(energy) is referred to as the power coupling factor (PCF) or energy coupling factor (ECF). The coupling factors have units of W/g-MW (J/g-MJ). The parameter is strongly influenced by the neutronic characteristics of the test specimen and test capsule. Determination of the PCF is initially achieved with a Monte Carlo neutronics model of the TREAT core and experiment design. The ECF is ultimately achieved via calibration runs with the neutronically equivilant test train at low powers.

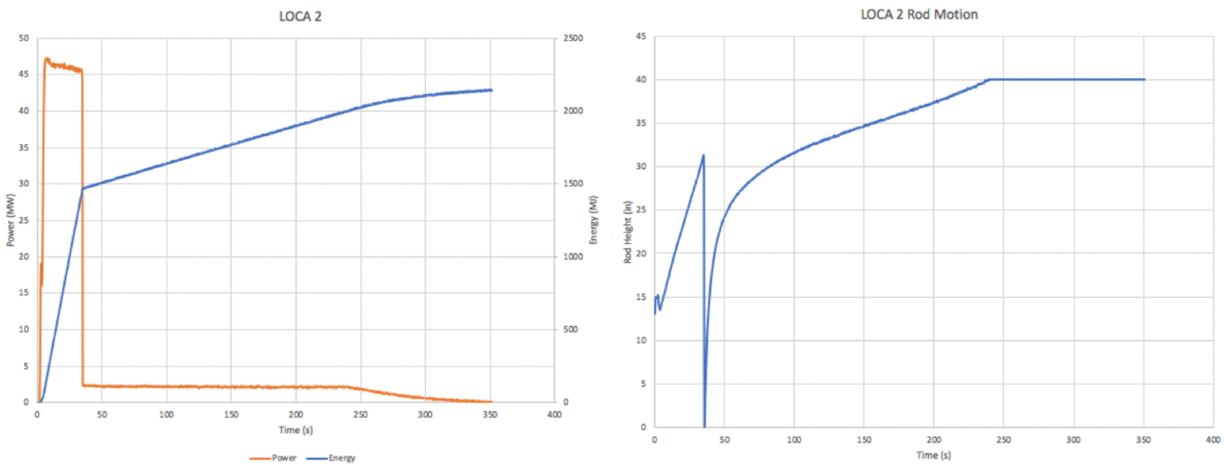

Figure 18. TREAT "shaped transient" simulating a LOCA power history.

While TREAT provides the unique nuclear heating of the ATF test specimens, the thermal hydraulic boundary conditions need to be supplied from unique experiment hardware that is designed independent from the reactor. To support transient testing of LWR fuel specimens including ATF, four separate experiment modules are being developed and deployed. Each involves a progressively higher level of sophistication in replicating the Thermal Hydraulic boundary conditions of an LWR. They names of these experiment modules or test trains are: (1) the Separate Effects Test Holder (SETH), (2) the static environment rodlet transient testing apparatus (SERTTA), (3) Super-SERTTA, and (4) the Transient Water Experiment Recirculating Loop (TWERL). Each system is described briefly in the following paragraphs.

3.3.1 Separate Effects Test Holder (SETH)

The SETH module is a static inert capsule that is capable of testing a nominally 6-inch test specimen with 4 inches of fissile material. The system lacks a purposeful heat rejection system but is useful for separate effects studies looking at the initial deposition and distribution of energy in the test specimen during a transient. The SETH module will fit inside a larger can called BUSTER (Broad Use Specimen Transient Experiment Rig) which will provide the safety containment boundary. Adaptations to the SETH capsule are also under investigation which include an artificial solid heat sink that can be used to generate prototypic radial temperature profiles in test specimens for a short duration transient. By manipulating the reactor power and electrical heater around the heat sink, a variety of transient radial temperature shapes can be created [39] (See Figure 19).

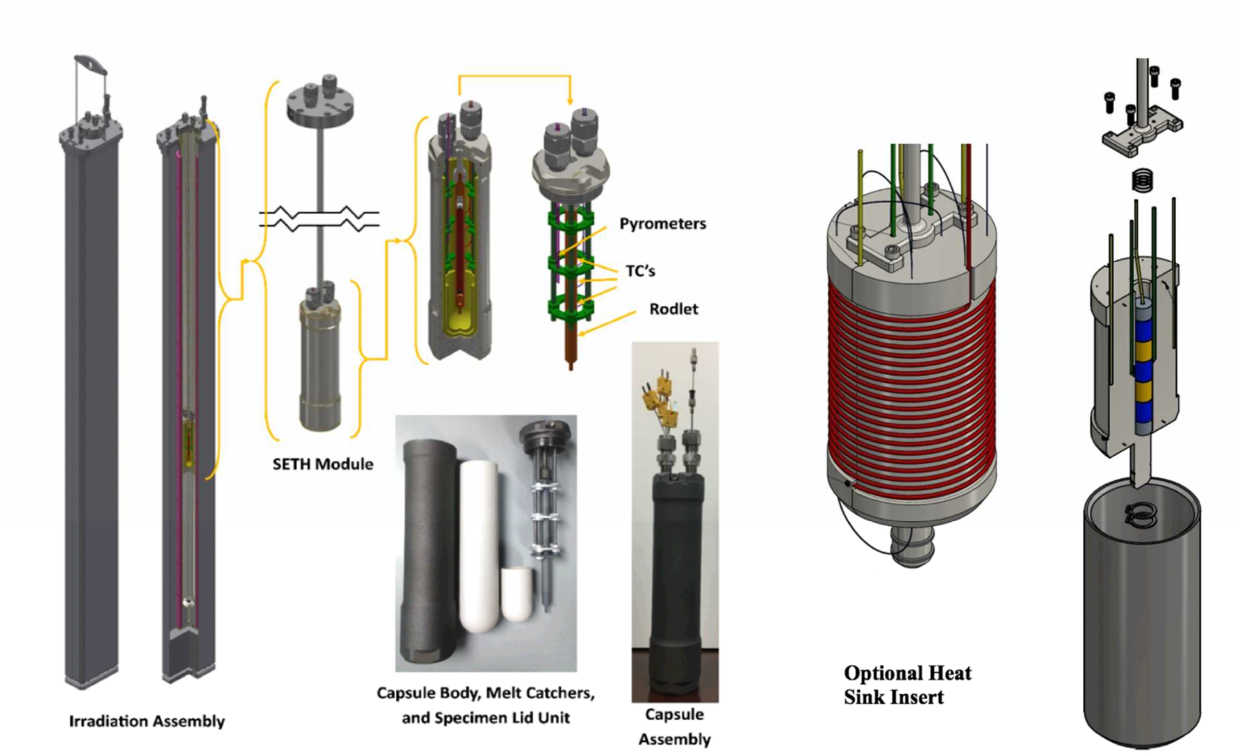

Figure 19. The SETH experiment module testing platform.

3.3.2 Static Environment Rodlet Transient Testing Apparatus (SERTTA)

The SERTTA Capsule is a static water environment test train capable of some elevated temperature and pressure testing. SERTTA also fits inside the BUSTER containment. Initial estimates for the temperature and pressure capability of MARCH-SERTTA are around 200°C and 3 MPa. The capsule is equipped with an expansion chamber above the test specimen to allow for the rapid vaporization of the water during the transient. This allows for some testing of material performance following DNB. A high temperature crucible is in placed at the bottom of the capsule to allow for partial or complete melting of the test specimen. Twelve 1 mm instrument leads penetrate the capsule and can be used to accommodate a variety of different instruments. Like the SETH module, this device is designed to accommodate 6-inch test specimens that contain a 4-inch stack of fissile material [40] (See Figure 20.)

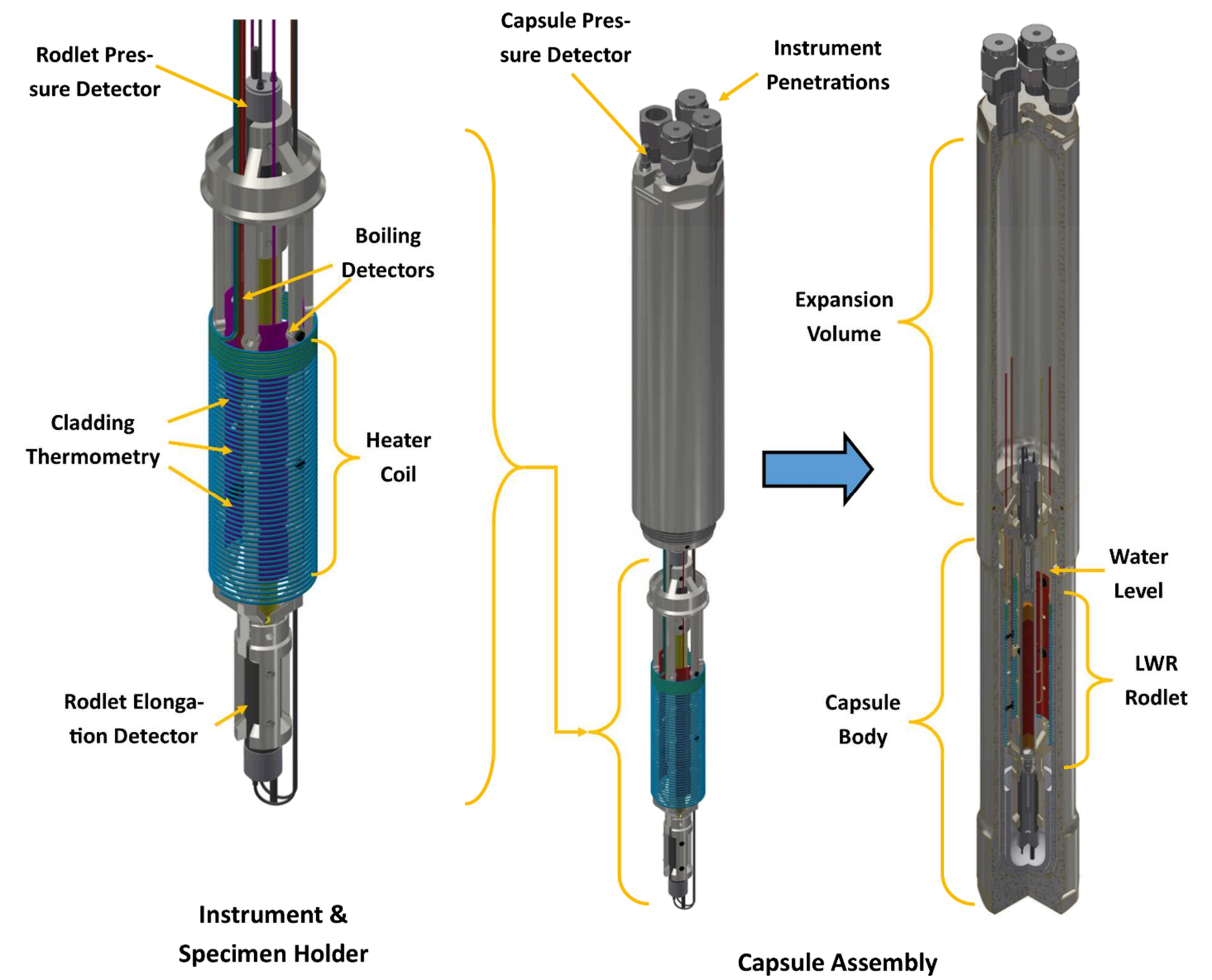

Figure 20. The SERTTA experiment module testing platform.

3.3.3 Super-SERTTA

An enlarged version of the SERTTA capsule is being developed called the Super-SERTTA device (Figure 21). This device is capable of fully replicating PWR and BWR temperature and pressures up to 16MPa and 300°C. Super-SERTTA has a large water volume with an annular flow channel so that some natural circulation in the device is achieved through activation of cartridge heaters which surround the test specimen and act as simulating neighbor rods. Super-SERTTA can accommodate larger test specimens initially designed to be 12 inches long. Some modifications may allow testing of even longer test specimens. Super-SERTTA is equipped with a large expansion tank that can serve as a blowdown system when simulating a LOCA transient or as a pressure relief system when simulating an RIA transient. A spray line is in the expansion tank to condense high temperature steam enabling the system to blowdown from PWR pressures to atmospheric pressure. Additionally, a reflood line allows for simulation of the reflood stage in the LOCA transient at various refill rates.

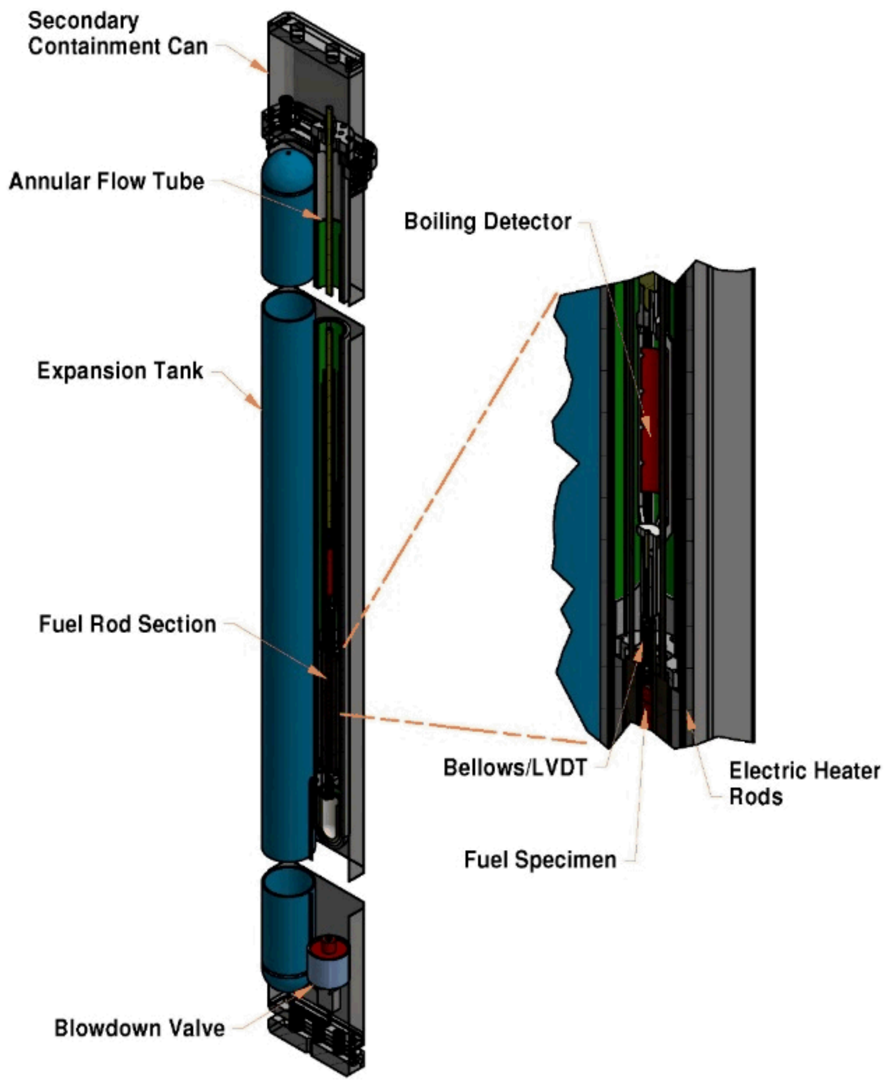

Figure 21. Super-SERTTA experiment module testing platform.

3.3.4 Transient Water Experiment Recirculating Loop (TWERL)

To fully achieve LWR testing conditions, an experiment device with a forced convection capability is being developed called TWERL. TWERL will be capable of simulating not only the temperature and pressure conditions of LWRs but also the coolant flow rates enabling fully prototypic heating of the test samples. TWERL is currently in the early stages of conceptual design (Figure 22) and a value engineering study is being conducted to determine if this capability can be achieved with a compact self-contained loop or if a facility scale system loop with ex-core support equipment will be required. This will largely depend on the need for testing fuel in a 3x3 bundle configuration or if smaller bundles, such as a 5 rod cruciform, can meet the testing needs. Test samples in TWERL will be able to be up to the full length of the active TREAT core approximately 3 feet.

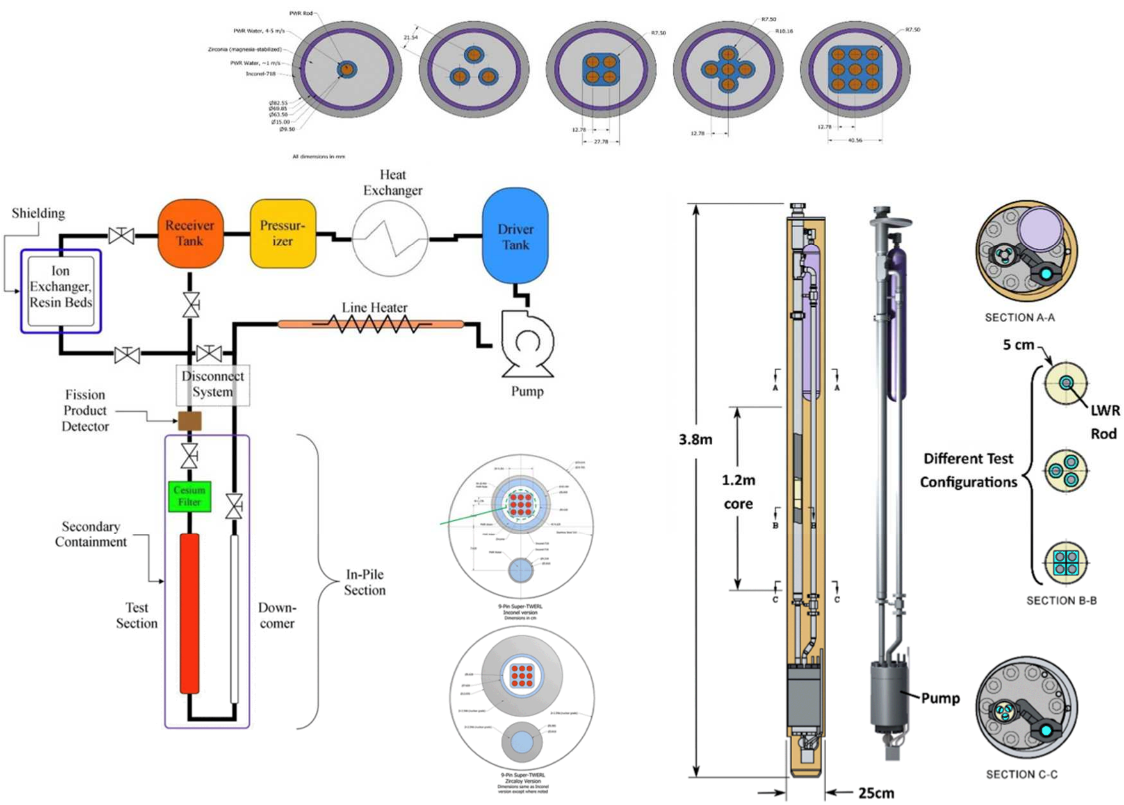

Figure 22. Design concepts for the TWERL experiment module testing platform.

4 INTEGRAL TESTING OF ATF

The principal aim of the fuel safety research program is to support the three U.S. nuclear fuel vendors in their development of their ATF concepts. As stated in Section 1, each vendor currently has or plans to have by the close of 2019 LTAs of their ATF concepts under irradiation in a commercial reactor. It is also planned that coated cladding and doped fuel designs be ready for batch scale implementation by 2023, and that full cores will be in place by 2026. Due to cost and schedule limitations, initially the integral ATF testing program at TREAT will be limited to coated cladding and doped fuel designs. More advanced designs will be considered as opportunities become. Presently, three test series are being proposed at the TREAT facility to support the licensing of these concepts and to realize the benefits of their enhanced safety performance in DBA and AOO transients. These plan to cover the spectrum of transient scenarios as well as levels of experiment complexity and integration. This will enable a sufficient understanding of the ATF transient behaviors to support licensing of the concepts. These test campaigns include:

1. An Integral RIA Test Series
2. An Integral LOCA Test Series
3. A bundle based PCM Test Series.

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 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, ATF LTA rods will be shipped to INL at 3 different burnup levels corresponding with the completion of 1, 2 and 3 cycles of commercial irradiation. 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 threshold as a result of the irradiation history.

Initially, INL through its AFC programmatic budget, will fund a demonstration or commissioning test for each test series with fresh UO_2-Zircaloy material. INL will also fund testing of previously irradiated (high burnup) UO_2-Zircaloy materials to demonstrate the ability to execute the tests on previously irradiated fuel. Both the fresh UO2-Zircaloy and previously irradiated UO_2-Zircaloy tests will demonstrate the suitability of these tests for developing nuclear fuel safety criteria. The behavior of UO_2-Zircaloy in accident scenarios has been rigorously studied through past decades through in-pile experiments at SPERT, LOFT, PBF, Halden (Norway), CABRI (France), and NSRR (Japan). The TREAT tests on UO_2-Zircaloy will validate these test methods through comparison with the historical database. After the demonstration tests with UO_2-Zircaloy, each fuel vendor through their FOA budget will be responsible for funding the test program for their ATF design and will be responsible for supplying both fresh and previously irradiated materials for the test series. Each fuel vendor has the flexibility to scope the test series based on their specific fuel design and specific testing needs.

4.1 ATF-RIA Test Series

The aim of the ATF-RIA test series is to study the performance of a given ATF fuel design in a prototypic design basis RIA. The tests will evaluate transient fuel behaviors that occur over short durations with high temperatures and high temperature ramp rates. The goal is to develop thresholds for fuel failure, radiological release, and maintenance of coolable geometry applicable to a given ATF design.

4.1.1 Scope

The test will initially be conducted in the SERTTA test capsule. Later tests may use more complex vehicles as they become available. Tests will be initiated from either a hot zero power condition or a cold zero power condition depending on the applicability of the design to PWRs or BWRs respectively. A prompt critical pulse will be initiated and will be subsequently clipped so that a nominally gaussian pulse is released. The selection of the initial step insertion and the clipping time of the transient will be selected to target a given energy release with a minimum pulse width. The energy released in the gaussian pulse will be systematically increased to create a more severe transient. Examples of energy release targets for fresh fuel tests may by 500, 700, 900, 1100, and 1300 J/g respectively. The specific values will be identified in the test plans for each ATF concept; however, it is envisioned that at least 5 transients will be conducted for the fresh fuel tests. Tests on previously irradiated ATF materials will be conducted near the hypothesized thresholds developed from the fresh fuel tests as well as any relevant out of pile testing data. As INL implements enhancements at TREAT for narrowing the pulse width of TREAT, these enhacements will be utilized to decrease the pulse width to be more representative of PWR CREAs and BWR CRDAs respectively. These tests will also be the first to utilize previously irradiated material, as such scope and funding are included to develop the hot cell infrastructure necessary for a baseline refabrication capability for the irradiated rods.

4.1.2 Schedule

Major program milestones for the test series are given below.

GFY - 2019

- Demonstration (Commissioning) test with fresh UO_2-Zircaloy
- Complete preliminary design of hot-cell refabrication system

GFY – 2020

- Continue with fresh UO_2-Zircaloy Tests, and conduct PIE
- Begin testing of fresh (un-irradiated) ATF designs
- Complete design and installation of hot-cell refabrication system and conduct a demonstration test with previously irradiated UO_2-Zircaloy

GFY – 2021

- Continue testing fresh ATF designs and conduct PIE
- Continue testing high burnup UO_2-Zircaloy and conduct PIE
- Shipment of low burnup (first cycle) ATF material from LTA to INL

GFY – 2022

- Begin testing of low burnup (first cycle) ATF

- Shipment of medium burnup (second cycle) ATF to INL

GFY – 2023

- Continue testing low burnup ATF and conduct PIE
- Begin testing of medium burnup ATF and conduct PIE

GFY – 2024

- Shipment of high burnup (three cycle) ATF to INL
- Begin testing of high burnup ATF.

4.1.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. Breakout is provided between the AFC programmatic contribution and the contribution expected from each fuel vendor to conduct their testing. The total budget includes the AFC programmatic budget and three times the Vendor FOA budget, as it is assumed all 3 vendor teams will participate.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$3000	$1500	$500	—	—
Vendor FOA	$1000	$1000	$1500	$1500	$1500
Total	**$6000**	**$4500**	**$5000**	**$4500**	**$4500**

4.2 ATF-LOCA Test Series

The aim of the ATF-LOCA test series is to study the performance of a given ATF concept in a design basis LOCA event. These tests will evaluate transient fuel behaviors that occur during longer time at temperature and with high pressure differentials across the cladding. The goal is to develop thresholds for fuel failure, radiological release, and maintenance of coolable geometry applicable to a given ATF design.

4.2.1 Scope

Tests will initially be conducted in the Super-SERTTA capsule. Later testing may take place in more sophisticated vehicles as they become available. TREAT will be operated in a shaped transient mode to replicate a period of steady state operations prior to initiation of the LOCA transient. After the initiation of the blowdown, the reactor power and adjacent cartridge heaters will be manipulated is such a way as to simulate decay heat adiabatically increasing the fuel rod temperature to a prescribed peak temperature and maintaining that temperature for a period of time before initiating a quench. The severity of the transient will be systematically increased by increasing the peak temperature of the transient. The time at peak temperature may also be increased within the limitations of TREAT's shaped transient capability of 2500 MJ total energy release. Peak temperature targets will be identified in the specific test plans; however, they may include 900, 1000, 1100, 1200, and 1300 °C. As with the RIA test series, it is envisioned that a minimum of 5 tests will be conducted with fresh fuel. Later tests on previously irradiated materials will be more focused on specific transition points identified in the fresh fuel tests as well as any applicable out of pile test data.

4.2.2 Schedule

Major program milestones for the test series are given below.

GFY - 2020

- Demonstration (Commissioning) test with fresh UO_2-Zircaloy

GFY – 2021

- Continue testing with fresh UO2-Zircaloy, and conduct PIE
- Begin testing of fresh (un-irradiated) ATF designs
- Demonstration test with high burnup UO2-Zircaloy

GFY – 2022

- Continue testing fresh ATF designs and conduct PIE
- Continue testing high burnup UO2-Zircaloy and conduct PIE
- Begin testing of low burnup ATF designs

GFY – 2023

- Continue testing of low burnup ATF and conduct PIE
- Begin testing of medium burnup ATF

GFY – 2024

- Continue testing of medium burnup ATF and conduct PIE
- Begin testing of high burnup ATF.

4.2.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. Breakout is provided between the AFC programmatic contribution and the contribution expected from each fuel vendor to conduct their testing. The total budget includes the AFC programmatic budget and three times the Vendor FOA budget, as it is assumed all 3 vendor teams will participate.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$2000	$2500	$1000	$500	—
Vendor FOA	—	$1000	$1500	$1500	$1500
Total	**$2000**	**$5500**	**$5500**	**$5000**	**$4500**

4.3 ATF-PCM Test Series

The aim of the ATF-PCM test series is to fill in any gaps in understanding of transient fuel behavior that may exist from the LOCA and RIA test series. The test will also take place in a bundle like geometry and will identify any unique phenomenon resulting from rod to rod interactions. Peak temperatures expected in the transient are intendent to be between those of the RIA and LOCA transient as are the duration of the transients. The goal of the test will be to develop thresholds for fuel damage, fuel failure, and maintenance of coolable geometry for a given ATF design in a generalized PCM event.

4.3.1 Scope

The tests will take place in the TWERL capsule. Studies are currently underway at INL to determine if a full 9 rod bundle will be used in these tests or if a 5 rod bundle will be used in a cruciform configuration. TREAT will be operated in a shaped transient mode to simulate a rod bundle operating at steady state power. The pump power in the test train will be rolled back to create a power cooling mismatch scenario where the top part of the fuel bundle experiences a boiling crisis. Reactor power will remain constant throughout the test. The test will be terminated by either increasing the pump power or when TREAT exceeds its 2500 MJ energy limit and the bundle power is reduced. Either outcome will result in the boiling crisis ending and the rods being quenched. The severity of the transient will be systematically increased by increasing the steady state linear power in the fuel rods, thus increasing their temperature in the stable film boiling state. Example tests could include tests at 30, or 40 kW/m respectively. Specific power targets will be identified in the PCM test plans. Tests with previously irradiated fuel will be conducted with only the center pin in either the 5 or 9 rod bundle being previously irradiated, the surrounding rods will still be comprised of fresh fuel materials that are the same nominal design as the irradiated pin.

4.3.2 Schedule

Major program milestones for the test series are given below.

GFY - 2020

- Complete Conceptual Design of TWERL

GFY - 2021

- Complete Preliminary Design of TWERL

GFY - 2022

- Demonstration (Commissioning) test with Fresh UO_2-Zircaloy

GFY – 2023

- Complete Fresh UO_2-Zircaloy tests, conduct PIE
- Begin testing of fresh (un-irradiated) ATF designs
- Demonstration test with previously irradiated UO_2-Zircaloy

GFY – 2024

- Continue tests with previously irradiated UO_2-Zircaloy, conduct PIE
- Begin testing of previously irradiated ATF.

4.3.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. Breakout is provided between the AFC programmatic contribution and the contribution expected from each fuel vendor to conduct their testing. The total budget includes the AFC programmatic budget and three times the Vendor FOA budget, as it is assumed all 3 vendor teams will participate.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	\$500	\$1000	\$3000	\$3000	\$2000
Vendor FOA	—	—	—	\$1000	\$1500
Total	**\$500**	**\$1500**	**\$3000**	**\$6000**	**\$6500**

5 ATF Focused Effects Testing

While the priority for the fuel safety research program is to develop the unique infrastructure needed to directly support the fuel vendors testing needs, it is also an identified objective of the national laboratory research program to conduct independent testing that generically supports the goals of the ATF program. The fuel safety research program has identified 5 strategic focused effects test series that can be used in concert with the fuel vendor efforts to provide important data which can be used to understand ATF transient behavior. These include:

1. A test series to study the effect of pulse width on RIA PCMI failure thresholds
2. A test series to generically study performance of chrome coated cladding in various time at temperature transients
3. A test series to generically study the performance of non-oxide fuels in ceramic composite claddings in RIA transients
4. A test series to study fuel fragmentation and transient fission gas release thresholds in regular and doped UO_2 fuels at different burnup levels
5. A test series to study fuel coolant interaction and the magnitude of pressure pulses generated in these events.

These tests will be funded entirely from the AFC program budget and, in specific cases, through collaborations with universities, industry partners, and international research organizations. It is recognized that, if DOE continues to fund all three fuel vendors in their development of ATF, the test programs described in Section 4 alone require a significant amount of the resources available at INL for conducting TREAT tests. The test series described in this section will have a secondary priority to the integral tests. However, these tests will move forward with the identified milestones so that the fuel safety research program can continue to make credible contributions to achieving the ATF objectives in the event there are challenges or delays with executing the vendor tests. It is also recognized that, as the demand for transient testing grows, efficiencies will be realized, and additional resources made available. It is the goal of these focused effects tests to be ready to take advantage of these opportunities. Additional focused effects tests may be proposed and added to this test plan in subsequent revisions as testing needs are identified.

5.1 Pulse Width Effects on PCMI Test Series

The pulse width of an RIA transient is thought to be a significant variable affecting the vulnerability of a given fuel concept to withstand PCMI failure. While the current NRC regulatory guides [15] do not take into account pulse width on PCMI failure thresholds in RIAs, intuitively transients with shorter pulse widths will result in pellet cladding interaction at lower temperatures and higher strain rates, both of which should make the cladding more vulnerable to PCMI failure. Most of the recent RIA testing on high burnup fuel has taken place in the CABRI and NSRR reactors which have much narrower pulse widths (4-20ms) than typical LWRs (30-80ms). As such, results from these tests are conservative when comparing to an actual LWR transient. TREAT has a longer pulse width (90-200ms) and, as such, test data may be non-conservative when addressing PCMI failure modes. The aim of this test series is to attempt to identify and quantify any dependence of pulse width on PCMI failure thresholds.

5.1.1 Scope

The tests will take place in the SERTTA capsule with pre-hydrided zircaloy cladding and UO_2 pellets with a reduced pellet cladding gap. Test data from the previously irradiated UO_2-Zircaloy RIA test described in Section 4.1 will also be used in the analysis. The pre-hydrided cladding will simulate the principal environmental degradation mechanism of Zircaloy cladding thought to contribute to PCMI failure. The tests will be executed with prompt gaussian pulses at TREAT's nominal minimum pulse width (~90ms) and energy releases designed to cause a 2% cladding hoop strain based on the thermal expansion of the pellet. The test will involve two different cladding hydrogen compositions and will be conducted at both cold and hot coolant conditions. The tests will be repeated at either a partner transient test facility such as NSRR in Japan or ACRR at Sandia National Laboratory, or in TREAT when the He-3 clipping system is installed at a narrower pulse width.

5.1.2 Schedule

Major program milestones for the test series are given below.

<u>*GFY - 2020*</u>

- Initiate tests series at TREAT with a nominal TREAT pulse width

<u>*GFY – 2021*</u>

- Complete TREAT test series and conduct PIE

<u>*GFY – 2022*</u>

- Repeat tests at a facility with a narrower pulse width and conduct PIE.

5.1.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. The budget listed here does not include funding for conducting tests as the partner facility. If a strategic partner such as NSRR or ACRR is not identified with an independent source of funding, the cost estimate will need to be updated to include testing at TREAT with the He-3 clipping system.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$750	$500	$500	—	—

5.2 Generic Chrome Coated Cladding Test Series

Chrome coated claddings are thought to be the leading near-term ATF concept which is to be deployed in batch reloads by 2023 and make up full LWR cores by 2026. Chrome coated claddings are thought to have improved swelling, rupture, and time at temperature behavior over standard Zircaloy claddings. The aim of this study is to look at coatings from a generic point of view to attempt to quantify these benefits and to identify what features (density, thickness, grain morphology etc.) are required for the coatings to provide these benefits in DBA scenarios.

5.2.1 Scope

Through partnerships with universities that have coating experience, coated Zr-4 and Zr-2 rodlets will be fabricated with varying application methods and at varying thicknesses. In addition to extensive out of pile testing and characterization, transient testing in a variety of post-DNB conditions will take place to replicate the cladding thermal mechanical conditions from internal nuclear heating. The test rods will be fabricated with varying internal pressures to evaluate the swelling and rupture behavior for different differential pressures. The in-pile transient testing will involve pulses and shaped transients that are tuned to provide different peak cladding temperatures.

5.2.2 Schedule

Major program milestones for the test series are given below.

GFY – 2019

- Develop university contracts to coat claddings

GFY - 2020

- Conduct out of pile mechanical characterizations of coated claddings

GFY – 2021

- Begin in-pile testing of coated claddings in transients that simulate post DNB behavior

GFY – 2022-2024

- Continue in-pile testing of coated claddings in transients that simulate post DNB behavior.

5.2.3 Budget

While cost estimates for the work scope have not been generated a programmatic funding target is presented below. The budget listed here is for the AFC programmatic contribution.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$500	$750	$750	$1500	$1500

5.3 Ceramic Composite Cladding with Non-Oxide Fuel Test Series

Revolutionary ATF designs have the potential to provide a significant benefit in both normal operations, design basis accidents, and beyond design basis accident conditions. These concepts generically consist of a Silicon Carbide ceramic matrix composite cladding (SiC-SiC) with a non-oxide fuel pellet such as a Uranium Silicide or Uranium Nitride. These concepts have potentially unique failure modes in RIA transients as a result of the fact that the fuel pellets may have a lower melting point than the cladding and that the ceramic claddings are stiff with limited ductility available in a PCMI event. It is the aim of this testing to evaluate these unique failure modes and to test different fuel design variants to optimize their performance in these conditions.

5.3.1 Scope

Initially, tests will take place in the SETH test capsule; however, testing will transition to the SERTTA capsule once available. Pulse type transients will be initiated with the goal of either causing fuel melting or transferring a desired hoop strain to the cladding as the result of PCMI based on the best available thermal mechanical data for the pellet and cladding materials.

5.3.2 Schedule

Major program milestones for the test series are given below.

GFY – 2019

- Conduct RIA Testing of U_3Si_2 in SiC-SiC cladding in the SETH Capsule

GFY - 2020

- Conduct RIA Testing of U_3Si_2 in SiC-SiC cladding in the SERTTA Capsule

GFY – 2021-2024

- Continue RIA Testing of various fuel designs involving either U_3Si_2 or UN in various SiC-SiC cladding architectures in the SERTTA Capsule.

5.3.3 Budget

While cost estimates for the work scope have not been generated a programmatic funding target is presented below. The budget listed here is for the AFC programmatic contribution.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$500	$500	$500	$500	$1000

5.4 Fuel Fragmentation and Transient Fission Gas Release Test Series

During both LOCA, and RIA transients, the fuel pellet can experience dramatic and sudden changes in its radial temperature profile resulting in high thermal stresses and fission gas bubble pressures which can lead to fuel fragmentation and transient fission gas release. The aim of these tests is to quantify the thresholds for these events with fuel at various burnup levels and to understand any changes to thresholds as the result of large grain or doped UO_2.

5.4.1 Scope

The tests will involve small samples of previously irradiated fuel in which the cladding contact pressure is removed by cutting a slit in the side of the cladding. The test will take place in a modified version of the SETH capsule which provides independent control of a heat sink temperature and the capsule's hydrostatic pressure. It will also include fission gas release monitoring capabilities. RIA tests will start from a zero-power condition and will involve a prompt critical pulse of a specific energy release. LOCA tests start from a nominal steady state power, and then reactor power will be reduced, and heat sink temperature increased to flatten out the radial temperature profile; after this, reactor power and heat sink temperature will be increased in concert to simulate adiabatic heating of the pellet from decay heat. Thresholds will be developed for transient fission gas release, and fuel fragmentation for both RIA and LOCA conditions as functions of burnup, hydrostatic pressure, and fuel temperature. The project will also include synergistic furnace tests and advanced microstructural characterization of high burnup fuel to support the findings from the in-pile experiments and to test various hypothesis for fuel fragmentation and transient fission gas release.

5.4.2 Schedule

Major program milestones for the test series are given below.

GFY – 2020

- Complete Design modifications to SETH to include heat sink, hydrostatic pressure capability, and fission gas release monitoring capability
- Conduct furnace tests and microstructural studies of high burnup UO_2

GFY - 2021

- Conduct in-pile tests in modified SETH with high burnup UO_2
- Continue furnace tests and microstructural studies of high burnup UO_2

GFY – 2022-2024

- Continue in-pile tests in modified SETH with UO_2 at different burnup levels
- Conduct in-pile tests in modified SETH with doped UO_2 at different burnup levels
- Conduct furnace tests and microstructural studies of UO_2 and doped UO_2 at various burnup levels.

5.4.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. The budget listed here is for the AFC programmatic contribution.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	$750	$750	$1000	$1500	$1500

5.5 Fuel Coolant Interaction (FCI) Test Series

During an RIA transient where the cladding fails and the fuel fragments and/or melts, there is the possibility for large pressure pulses to be generated from the result of vapor generation and collapse as the result of fragmented or molten fuel coolant interaction. The inability to accurately predict the size of these pressure pulses limits most LWRs to stay below the fuel failure thresholds for RIA events with high burnup fuel. Intuitively, the size of the pressure pulse should be related to the size of the fuel fragments and the peak temperature of the fragments. The aim of this study will be to better understand the size of the pressure pulses generated as the result of FCI.

5.5.1 Scope

The test series will involve fuel fragments of specified sizes being submerged in water and then subject to a prompt power pulse in TREAT. The tests will take place in a modified SETH capsule that is capable of measuring the pressure pulse. The severity of the transient will be increased by conducting tests with finer fuel particle sizes and higher transient energy releases, resulting in higher particle temperatures. Later tests with prototypic water-logged pins will take place in the SERTTA capsule. For these tests, an engineered defect will be created in the cladding so that water in the SERTTA capsule fills the fuel cladding gap. A prompt power pulse in TREAT will then be executed that is representative of a PWR CREA or a BWR CRDA.

5.5.2 Schedule

Major program milestones for the test series are given below.

GFY – 2021

- Begin design modifications to SETH to include capability to test fuel particles in water

GFY - 2022

- Conduct FCI SETH irradiations
- Analyze SERTTA to be capable of testing water-logged pin

GFY – 2023

- Continue FCI SETH irradiations
- Conduct SERTTA irradiation with an unirradiated water-logged fuel pin

GFY – 2024

- Continue FCI SETH irradiations
- Conduct SERTTA irradiation with a high burnup water-logged fuel pin.

5.5.3 Budget

While cost estimates for the work scope have not been generated, a programmatic funding target is presented below. The budget listed here is for the AFC programmatic contribution.

	GFY – 2020	GFY – 2021	GFY – 2022	GFY – 2023	GFY – 2024
AFC Program	—	$500	$750	$1000	$2000

6 PROGRAM ROADMAP

Key milestones by year for this fiscal year and the next 5 fiscal years are presented in Table 1 below. Not all milestones or activities are represented, only the most notable and impactful from each test series is shown.

Table 1. ATF safety research program key milestones.

	Integral ATF Test Series	Focused Effects ATF Test Series
2019	• Demonstration (Commissioning) RIA test with fresh UO_2-Zircaloy • Complete preliminary design of hot-cell refabrication system.	• Conduct RIA Testing of U_3Si_2 in SiC-SiC cladding in the SETH Capsule
2020	• Begin RIA testing of fresh (un-irradiated) ATF designs • Complete design and installation of hot-cell refabrication system and conduct a demonstration test with previously irradiated UO_2-Zircaloy • Demonstration (Commissioning) LOCA test with fresh UO_2-Zircaloy • Complete Conceptual Design of TWERL	• Conduct PCMI RIA test at TREAT with pre-hydrided Zircaloy to characterize pulse width effect
2021	• Shipment of low burnup (first cycle) ATF material from LTA to INL • Begin LOCA testing of fresh (un-irradiated) ATF designs • Demonstration LOCA test with high burnup UO_2-Zircaloy • Complete Preliminary Design of TWERL	• Begin in-pile testing of coated claddings in transients that simulate post DNB behavior • Conduct Fuel Fragmentation and Transient Fission Gas release tests with high burnup UO_2
2022	• Begin RIA testing of low burnup (first cycle) ATF • Begin LOCA testing of low burnup ATF designs • Demonstration (Commissioning) bundle PCM test with Fresh UO_2-Zircaloy • Shipment of medium burnup (second cycle) ATF to INL	• Conduct Fuel Fragmentation and Transient Fission Gas release tests with doped UO_2
2023	• Begin RIA testing of medium burnup ATF • Begin LOCA testing of medium burnup ATF • Begin bundle PCM testing of fresh (un-irradiated) ATF designs • Demonstration bundle PCM test with previously irradiated UO_2-Zircaloy	• Conduct SERTTA irradiation with an unirradiated water-logged fuel pin.

	Integral ATF Test Series	Focused Effects ATF Test Series
2024	• Shipment of high burnup (three cycle) ATF to INL • Begin RIA testing of high burnup ATF • Begin LOCA testing of high burnup ATF • Begin bundle PCM testing of previously irradiated ATF	• Conduct SERTTA irradiation with a high burnup water-logged fuel pin.

A five-year budget is provided in Table 2 below to fund the activities described in this plan. The budgets are for programmatic planning purposes and do not represent cost estimates required to complete specific scope items. They will be adjusted on a year by year basis, as necessary, based on execution of actual work scope and will evolve to meet programmatic needs and priorities.

Table 2. ATF fuel safety research programmatic budget table.

ATF Fuel Safety Research 5 Year Funding Profile (Numbers in $1000)					
	GFY - 2020	GFY - 2021	GFY - 2022	GFY - 2023	GFY - 2024
ATF - RIA	**$6,000**	**$4,500**	**$5,000**	**$4,500**	**$4,500**
AFC Program	$ 3,000	$1,500	$ 500	—	—
Individual Vendor Contribution	$ 1,000	$1,000	$1,500	$1,500	$1,500
ATF - LOCA	**$ 2,000**	**$5,500**	**$5,500**	**$5,000**	**$4,500**
AFC Program	$ 2,000	$2,500	$1,000	$ 500	—
Individual Vendor Contribution	—	$1,000	$1,500	$1,500	$1,500
ATF - PCM	**$ 500**	**$1,000**	**$3,000**	**$6,000**	**$6,500**
AFC Program	$500	$1,000	$3,000	$3,000	$2,000
Individual Vendor Contribution	—	—	—	$1,000	$1,500
ATF - Focused Effects	**$ 2,500**	**$3,000**	**$3,500**	**$4,500**	**$6,000**
AFC - PCMI	$750	$ 500	$ 500	—	—
AFC - Coated Cladding	$500	$ 750	$ 750	$1,500	$1,500
AFC – Non-Oxide fuel and SIC-SiC cladding	$500	$ 500	$ 500	$ 500	$1,000
AFC - FF and TFGR	**$750**	**$ 750**	**$1,000**	**$1,500**	**$1,500**
AFC - FCI	—	$ 500	$ 750	$1,000	$2,000
Total Program	**$11,000**	**$14,000**	**$17,000**	**$20,000**	**$ 21,500**
AFC Total	$8,000	$8,000	$8,000	$8,000	$8,000
Individual Vendor Total	$1,000	$2,000	$3,000	$4,000	$4,500

7 SUMMARY

The ATF program is on an aggressive schedule to meet key dates of 2023 and 2026 for the batch reload and full core implementation of coated cladding and doped UO_2 ATF concepts. Longer term concepts such as FeCrAl and SiC-SiC cladding and U_3Si_2 fuel pellets are on an equally aggressive schedule to be ready for batch reloads and full core implementation by 2026 and 2030 respectively. Licensing of a new fuel design requires knowledge of the known failure modes for the fuel design as well as knowledge of any phenomenon leading to a loss of coolable geometry. Knowledge of these thresholds and limits allows for the development of appropriate fuel safety criteria and establishment of analytical limits for the fuel designs performance and accident analysis. In order to develop these criteria, transient testing near the performance limits of the fuel design in a variety of AOO and DBA-like conditions at a variety of burnup levels is required. The ATF safety research program at INL is well suited to aid the three U.S. nuclear fuel vendors and the NRC in the task of safety licensing this new fuel designs by conducting prototypic in-pile transient testing at the TREAT reactor.

To support the goals and objectives of the ATF program, three integral testing campaigns and five crosscutting focused effects testing campaigns have been proposed in this program plan. The integral testing campaigns focus on replicating the key nuclear, thermal, and environmental conditions of design basis RIA, and LOCA transients, as well as generalized PCM transients. The focused effects testing campaigns look specifically at transient fuel behaviors likely to be most important to specific ATF concepts from a vendor neutral point of view. These tests look at transient fuel behavior related to pellet cladding mechanical interaction, coated cladding swelling rupture, and time at temperature behavior, non-oxide fuel melting and mechanical interaction with Silicon Carbide cladding, UO_2 and doped UO_2 fuel fragmentation and transient fission gas release, and finally fuel coolant interaction behavior. The AFC programmatic funding will be used to demonstrate the integral tests on existing UO_2-Zircaloy fuel designs both at the unirradiated and irradiated condition. This will allow the TREAT tests to connect back to the rich historical database of fuel safety testing conducted over the past decades. The AFC will also fund the execution of the crosscutting focused effects tests which generally support the ATF program. Each ATF vendor will be required to use funding from their FOA agreement to fund the integral testing of their specific design concept and may tailor these tests to meet their specific data objectives.

A roadmap for executing these tests in time to support the 2023 batch insertion and 2026 full core reload dates is given in this program plan. Low burnup testing is planned to be completed before the 2023 batch insertion date, and high burnup testing is initiated in 2024 and planned to be completed before the start of the 2026 full core implementation date, which is when ATF will need to be licensed for high burnup operation.

8 REFERENCES

[1] U.S. Department of Energy, “Roadmap: Development of Light Water Reactor Fuels with Enhanced Accident Tolerance Report to Congress,” 2015.

[2] Nuclear Energy Institute, “Accident Tolerant Fuel,” 2019. [Online]. Available: https://www.nei.org/advocacy/make-regulations-smarter/accident-tolerant-fuel.

[3] Nuclear Energy Agency, “Nuclear Fuel Safety Criteria Technical Review Second Edition NEA No. 7072,” 2012.

[4] C. Jensen, D. Wachs, N. Woolstenhulme, S. Hayes, G. Povirk, and K. Richardson, “Post-Halden Reactor Irradiation Testing for ATF : Preliminary Assessment and Recommendations,” 2018.

[5] Idaho National Laboratory, “Nuclear Materials Experiments Execution Process LWP-20700,” 2017.

[6] D. Kamerman and J. Payne, “Development of Database for Qualified Nuclear Data on Accident Tolerant Fuel Irradiations,” 2018.

[7] Nuclear Regulatory Comission, “NUREG-0800, Ch. 4.2, Rev. 3,” no. 301, pp. 33–36, 2007.

[8] C. B. Jensen, N. E. Woolstenhulme, and D. M. Wachs, “The TREAT Experiment Legacy Supporting LWR Fuel Technology,” in *Top Fuel 2018*, 2018, pp. 1–12.

[9] Nuclear Energy Agency, “Nuclear Fuel Behaviour Under Reactivity-initiated Accident (RIA) Conditions,” 2010.

[10] T. Fuketa, *Transient response of lwr fuels (ria)*, vol. 2. Elsevier Inc., 2012.

[11] Nuclear Regulatory Comission, “Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Bumup Fuel (NUREG/CR-6742),” 2001.

[12] Nuclear Regulatory Comission, “Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Bumup Fuel NUREG/CR-6744,” 2001.

[13] U.S. Atomic Energy Commission, “Regulatory Guide 1.77,” no. May, pp. 1–8, 1974.

[14] P. E. MacDonald, “Assessment of Light Water Reactor Fuel Damage During a Reactivity Initiated Accident,” in *CSNI Specialist Meeting on Safety Aspects of Fuel Behavior in Off-Normal and Accident Conditions*, 1980.

[15] Nuclear Regulatory Comission, “Draft Regulatory Guide DG-1327 Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents,” 2016.

[16] R. A. Lorsnz and W. Parker, “Final Report on the Second Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod cluster in TREAT,” 1972.

[17] Nuclear EnergyAngency, “Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions State-of-the-art Report,” 2009.

[18] F. Nagase, *Behavior of lwr fuel during loss-of-coolant accidents*, vol. 2. Elsevier Inc., 2012.

[19] Nuclear Energy Agency, “Report on Fuel Fragmentation Relocation Dispersal,” 2016.

[20] H. M. Chung and T. F. Kassner, “EMBRITTLEMENT CRITERIA FOR ZIRCALOY FUEL CLADDING APPLICABLE TO ACCIDENT SITUATIONS IN LIGHT-WATER REACTORS : SUMMARY REPORT NUREG/CR-1344,” 1980.

[21] M. Billone, Y. Yan, T. Burtseva, and R. Daum, "Cladding Embrittlement During Postulated Loss-of-Coolant Accidents NUREG/CR-6967," 2008.

[22] Nuclear Regulatory Comission, "Cladding Swelling and Rupture Models for LOCA Analysis NUREG-0630," no. April, 1980.

[23] F. Haggag, "Zircaloy Cladding Embrittlement Criteria : Comparison of In-Pile and Out-of-Pile Results NUREG/CR-2757," 1982.

[24] Nuclear Regulatory Comission, "Draft Regulatory Guide DG-1262 Testing for Post Quench Ductility," 2014.

[25] Nuclear Regulatory Comission, "Draft Regulatory Guide DG-1263 Establishing Analytical Limits for Zirconium-Based Alloy Cladding," 2012.

[26] P. E. Macdonald, A. S. Mehner, W. J. Quapp, Z. R. Martinson, and R. K. Mccardell, "RESPONSE OF UNIRRADIATED AND IRRADIATED POWER-COOLING-MISMATCH CONDITIONS," 1978.

[27] Nuclear Energy Agency, "State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels," 2018.

[28] K. A. Terrani, "Accident tolerant fuel cladding development: Promise, status, and challenges," *J. Nucl. Mater.*, vol. 501, pp. 13–30, 2018.

[29] C. Tang, M. Stueber, H. J. Seifert, and M. Steinbrueck, "Protective coatings on zirconium-based alloys as accident-Tolerant fuel (ATF) claddings," *Corros. Rev.*, vol. 35, no. 3, pp. 141–165, 2017.

[30] Z. Duan *et al.*, "Current status of materials development of nuclear fuel cladding tubes for light water reactors," *Nucl. Eng. Des.*, vol. 316, pp. 131–150, 2017.

[31] M. W. D. Cooper, C. R. Stanek, and D. A. Andersson, "The role of dopant charge state on defect chemistry and grain growth of doped UO 2," *Acta Mater.*, vol. 150, pp. 403–413, 2018.

[32] J. T. White, A. T. Nelson, J. T. Dunwoody, D. D. Byler, D. J. Safarik, and K. J. Mcclellan, "Thermophysical properties of U3Si2 to 1773 K," *J. Nucl. Mater.*, vol. 464, pp. 275–280, 2015.

[33] J. Brachet and A. E. Commission, "BEHAVIOR OF Cr-COATED M5 CLADDINGS DURING AND AFTER HIGH TEMPERATURE STEAM OXIDATION FROM 800 ° C UP TO 1500 ° C (LOSS-OF- COOLANT ACCIDENT & DESIGN EXTENSION CONDITIONS) BEHAVIOR OF Cr-COATED M5 CLADDINGS DURING AND AFTER HIGH TEMPERATURE STEAM OXIDATI," *TopFuel*, no. October, pp. 1–10, 2018.

[34] M. Dumerval, Q. Houmaire, J. C. Brachet, H. Palancher, J. Bischoff, and E. Pouillier, "BEHAVIOR OF CHROMIUM COATED M5 CLADDINGS UPON THERMAL RAMP TESTS UNDER INTERNAL PRESSURE (LOSS-OF-COOLANT ACCIDENT CONDITIONS)," *Top Fuel 2018*, pp. 1–8, 2018.

[35] C. P. Massey, K. A. Terrani, S. N. Dryepondt, and B. A. Pint, "Cladding burst behavior of Fe-based alloys under LOCA," *J. Nucl. Mater.*, vol. 470, pp. 128–138, 2016.

[36] K. G. Field, X. Hu, K. C. Littrell, Y. Yamamoto, and L. L. Snead, "Radiation tolerance of neutron-irradiated model Fe-Cr-Al alloys," *J. Nucl. Mater.*, vol. 465, pp. 746–755, 2015.

[37] C. Davis, "Application of a RELAP5-3D Point Kinetics Model to a Full-Slotted Core in TREAT ECAR-3907," 2018.

[38] J. D. Bess *et al.*, "Narrowing transient testing pulse widths to enhance LWR RIA experiment design in the TREAT facility," *Ann. Nucl. Energy*, vol. 124, pp. 548–571, 2019.

[39] N. Woolstenhulme *et al.*, "New Capabilities for In-Pile Separate Effects Tests in TREAT," in *ANS Annual Meeting Conference Proceedings*, 2018, pp. 2–4.

[40] N. Woolstenhulme, A. Chipman, D. Dempsey, C. Folsom, D. Kamerman, and S. Snow, "Summary Technical Report for Conceptual Design of MARCH-SERTTA INL/EXT-18-51912," 2018.