Advanced Fuels Campaign Plan to Support the Establishment of Cladding Time-at-Temperature Criterion

Nuclear Technology Research and Development

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ACRONYMS

AFC	Advanced Fuel Campaign
AOO	anticipated operational occurrence
ATF	accident-tolerant fuel
BWR	boiling water reactor
CHF	critical heat flux
DO	dryout
DNB	departure from nucleate boiling
LOCA	loss-of-coolant accident
LWR	light water reactor
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NRC	Nuclear Regulatory Commission
OLMCPR	operating limit minimum critical power ratio
PIE	post-irradiation examination
PWR	pressurized water reactor
pRXA	partially recrystallized
RIA	reactivity insertion accident
RXA	recrystallized
SMR	small modular reactor
SRA	stress relieved annealed
t@T	time-at-temperature

ADVANCED FUELS CAMPAIGN PLAN TO SUPPORT THE ESTABLISHMENT OF CLADDING TIME-AT-TEMPERATURE CRITERIA

1. INTRODUCTION

Most nuclear licensing frameworks worldwide use dryout (DO) or departure from nucleate boiling (DNB) criteria to assess fuel cladding's ability to transfer heat during transient conditions. Specifically, DO and DNB criteria are commonly used to evaluate cladding heat transfer during an anticipated operational occurrence (AOO) and postulated accident conditions, defining the point at which the critical heat flux (CHF) is exceeded. Fuel rods that exceed CHF—whether through DO or DNB—are considered to have failed, regardless of the duration or severity of the event [1,2]. This assumption is based on the significant change in coolant conditions that occurs when controlled boiling transitions to steam (in the case of DO) or when nucleate boiling transitions to film boiling (in the case of DNB) occurs. Both transitions create a vapor or steam film that insulates the hot fuel rod, severely diminishing heat transfer. As a result, the cladding experiences a rapid temperature increase. This rapid rise in temperature may degrade the cladding's material properties, resulting in potential failure or embrittlement if the cladding experiences a sufficiently high temperature for a sufficient duration.

In the U.S., the Nuclear Regulatory Commission (NRC) uses the DO and DNB criteria as conservative thresholds to prevent fuel failure when the CHF is exceeded [1]. However, because DO and DNB are thermal-hydraulic phenomena, they do not fully describe the actual impact on the performance of the fuel rod materials. As a result, they are inherently conservative as failure criteria. A more accurate cladding failure criterion would consider not only the rate of temperature increase and decrease, terminal temperature but also the duration of sustained high-temperature exposures. These factors are crucial for assessing how the cladding material properties evolve under transient conditions. Cladding mechanical property changes would then need to be taken into account when evaluating the continued safe operation of the fuel and, if necessary, post-transient handling of the fuel.

Many AOOs involve a reactor trip, which decreases the amount of heat that must be removed by the coolant. Although these AOOs are relatively rare—especially those with short durations in CHF conditions—international experience with reactor operation in DO conditions suggests that the current, strict DO criteria may be overly conservative [1-5]. Adopting a fuel integrity criterion that allows for short durations in CHF conditions could provide additional operational margins [2]. This would enable improvements in several areas, such as increased plant operational flexibility, fuel conditioning, quicker power level adjustments, enhanced fuel cycle economics with more efficient core designs, and better-optimized boiling water reactor (BWR) control rod sequencing. Generic assessments indicate that the value of these changes is on the order of $1M to $1.5M USD per cycle for BWRs [4]. Specifically, the development of a new limit to replace the current operating limit minimum critical power ratio (OLMCPR) is being explored for BWRs. Substituting the OLMCPR with a thermomechanical limit based on time-at-temperature (t@T) could offer sufficient margin to uprate many high-power-density plants that are currently constrained by OLMCPR. The potential benefits for pressurized water reactors (PWRs) are more complex and less straightforward. PWR AOOs are generally more severe than those in BWRs, and the effective use of additional margin may depend on a variety of factors. Nevertheless, PWRs could see a potential ~5% increase in power output or peaking factors, which, when combined with successful burnup extension and higher enrichment levels, can further enhance performance [4]. The Advanced Fuels Campaign (AFC) will prioritize the reactor type and cladding materials that have the highest impact, as a direct response to the President's Executive Orders. Lastly, light-water small modular reactors (SMRs) utilize either natural circulation to minimize equipment and maintenance costs or forced circulation. Natural circulation enhances the plant's safety during a station blackout event. However, it results in higher coolant temperatures, which reduce the margin to CHF and thus potentially lead to less efficient core designs. However, forced circulation reduces temperatures and may allow for additional margin to CHF. The development and integration of new criteria could significantly improve the ability of SMRs to design and operate plants more efficiently.

The AFC has identified the development and establishment of new, t@T criteria as a strategic goal for the program. Preliminary analysis has shown that during unlikely transients, such as locked rotor events in PWRs and turbine trips without bypass in BWRs, fuel cladding temperatures increase rapidly, potentially exceeding 1000°C before being quenched after several seconds. This brief temperature rise can induce changes in the fuel cladding microstructure and, consequently, its properties—referred to as *time-at-temperature effects*. The primary phenomena that occur during and following these transients include recovery of irradiation hardening, recrystallization, time- and spatially dependent cladding creep, potential time-independent plasticity, hydride dissolution, cladding α/β transition, high-temperature corrosion at the cladding outer diameter (OD), and high-temperature oxygen diffusion from the cladding inner diameter via the pellet bonding layer. Understanding each of these phenomena under various conditions is crucial to developing predictive capabilities for fuel behavior and properties. This knowledge is vital for ensuring the continued safe use and handling of fuel, as well as for establishing conservative criteria to predict potential fuel failure.

This project plan aims to identify specific data needs and tasks that support replacing the existing CHF criteria with a t@T criterion. Additionally, critical gaps beyond CHF must be investigated to ensure the continued safe use and handling of fuel. The licensing of the t@T methodology must be completed by the product's vendor; however, this plan does not outline all data needs and tasks required for licensing. Instead, it specifically focuses on the fundamental data needs that can be generated through testing and examination of both unirradiated and irradiated fuel specimens. The data required for licensing will be the responsibility of the vendors, and, therefore, the program must collaborate closely with them to collect the necessary data. These examinations and tests can be conducted by the national laboratories and universities.

Lastly, this document outlines the long-term strategy and high-level needs to fulfill the stakeholder requirements. The specific priorities and testing needs will be discussed through the Collaborative Research for Advanced Fuels Technology (CRAFT) initiative, as well as through collaborations with other programs such as the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program and the Nuclear Fuel Industry Research (NFIR) program.

2. PROJECT PLAN FOR DEVELOPING t@T CRITERIA TO SUPPORT FUEL REUSE AND FUEL HANDLING POST-CHF

The project plan for supporting fuel reuse after a CHF event is based on the assumption that cladding temperature must remain below the α/β transition, which varies depending on the alloy composition. Fuel handling post-CHF is primarily associated with post-transient embrittlement; therefore, the cladding temperatures may be permitted to exceed the α/β transition and possibly enter the β regime. Accordingly, two separate criteria will be used to support the fuel reuse case and the case in which the cladding is permitted to exceed the α/β transition and possibly enter the β regime. Additionally, fuel entering its first cycle of operation and approximately the first half of the second cycle has been identified as the limiting case. Its higher operating power in the event of flow loss or its higher reactivity and subsequent enthalpy rise in the case of a reactivity-driven event results, in both scenarios, in rapid temperature ramps on the order of 100°C/s. The temperature ranges and times differ between PWR and BWR fuel, with PWR events experiencing higher peak temperatures. Using fast temperature ramps—where the up- and down-ramp durations are shorter than the hold time at peak temperature—helps minimize annealing effects during the ramp phase and is preferred for modeling simplicity. Using this assumption, the following phenomena and material properties are important to consider.

- **Recovery (irradiation and thermal):** The thermal transient will heat the cladding to high temperatures, maintaining those temperatures for brief periods. During this time, irradiation defects as well as defects induced from thermo-mechanical processing and deformation that occurs prior to the t@T event will have the opportunity to diffuse and ultimately annihilate within the material. The uncertainty lies in the post-transient mechanical properties: How many defects (i.e., irradiation defects, lines defects) will be recovered, is there any remaining residual thermal stresses generated by the transient, and what will the cladding's mechanical properties be after the transient?

- **Recrystallization:** The heat treatment and fabrication process of the Zircaloy cladding alloy can result in different microstructures, such as stress-relief annealed (SRA), partially recrystallized annealed (pRXA), or recrystallized annealed (RXA). The thermal transient may reach temperatures and times that exceed the material's fabrication conditions, thereby altering the cladding microstructure. This change in microstructure may affect the mechanical properties, potentially leading to less favorable fuel performance after the transient. Specifically, recrystallization can substantially alter the material's texture and grain size, which will significantly affect the strength (yield, creep) of the cladding material along the hoop and axial direction.

- **Yield Stress:** A clear failure criterion for the cladding would be if a t@T transient resulted in the stress exceeding the cladding's yield strength or modifying the yield strength of the pristine sample. Indeed, the risk associated with such yield stress lies in that mechanical loads imposed on the cladding may exceed either the yield or the creep strength. Yield stress is typically measured at operating temperatures (<400°C), so it is essential to gather sufficient data to develop a model to analyze cladding performance. Additionally, the yield stress will change as a function of irradiation dose, irradiation and thermal recovery, and recrystallization. While these changes depend on the fabrication process, they must be considered when collecting data for such an analysis.

- **Creep:** Thermal creep is usually a minor contributor to overall deformation during steady-state operation and is primarily significant under high-temperature conditions (e.g., during a loss-of-coolant accident). However, most creep data focuses on radial deformation, whereas an AOO event could lead to axial creep due to high temperatures and system pressures. Additionally, CHF conditions are expected to be highly localized (both axially and azimuthally) and could even lead to collapse of the cladding. This raises the question: How might localized creep mismatch or collapse impact post-transient fuel performance? Will the cladding creep more rapidly in certain areas, increasing the likelihood of failure due to pellet–cladding interaction? The safety significance of this behavior remains unclear, so further investigation is warranted.

- **Second Phase Particle (SPP) Growth and Distribution and Post-Transient Oxidation:** SPPs are incorporated during the alloying and heat treatment processes to mitigate oxidation, corrosion, and hydrogen pickup. During rapid thermal transients, these particles may either grow or dissolve due to the transient conditions, potentially altering the alloy's post-transient oxidation behavior. As a result, the oxide layer could grow beyond the thickness limits specified in the technical regulations.

- **Oxygen Uptake and Hydrogen Pickup:** Embrittlement in zirconium alloys has been a well-established concern for many decades. A specific issue during high-temperature thermal transients is the uptake and subsequent embrittlement that can occur during a thermal excursion. This phenomenon is particularly relevant in scenarios such as power–cooling mismatches, reactivity insertion accidents, and loss-of-coolant accidents. While this issue has been extensively studied, it remains unclear whether the previous findings are applicable to modern zirconium alloys. Therefore, confirmatory studies are necessary to assess the relevance of past data, and additional research may be needed to develop criteria aimed at mitigating potential dose consequences.

The experimental testing for re-use and post-transient fuel handling will follow a phased approach, as illustrated in Figure 1. Phase 1 comprises a series of targeted separate effects tests designed to support the development of an initial material model. The goal of these tests will be to characterize the evolution of material properties and establish a failure criterion. This phase also involves a high-temperature testing campaign to evaluate the mechanical behavior of both unirradiated and irradiated materials. A subsequent, long-term effort will address remaining uncertainties through extended autoclave and re-irradiation testing. In parallel, a modeling and simulation component is essential to translate material-level test data into predictions of transient performance. In areas where experimental testing may be limited or impractical, mechanistic model development will play a critical role in capturing relevant material behavior. Mechanistic model development will be owned by the NEAMS program, while AFC will support the experimental testing. Cross-collaboration between the two programs is required to validate models and codes as well as to identify high-impact data or gaps within the data. A detailed summary of the testing requirements is provided in Table 1, which outlines the range of experiments needed to support the development of a t@T criterion.

Table *2* identifies key modeling and simulation gaps and highlights opportunities for coordination with NEAMS to minimize redundancy and maximize programmatic impact.

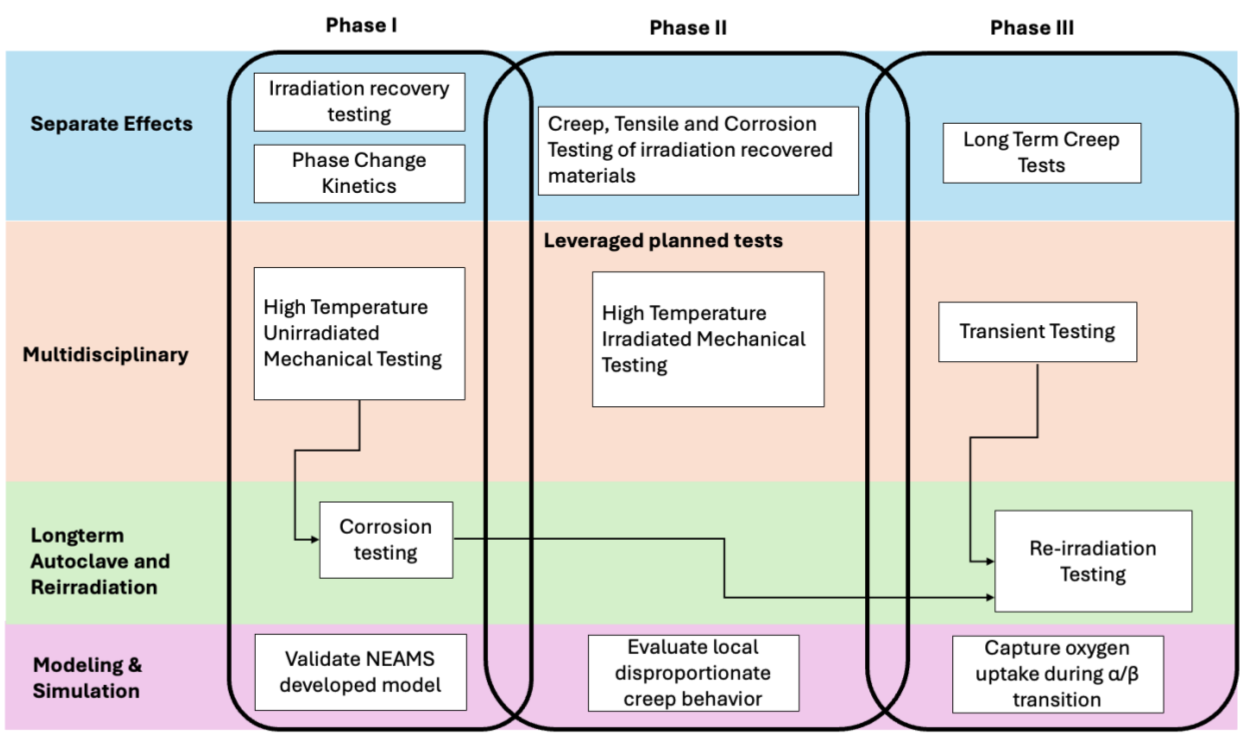

Figure 1 Phased approach to support the development of t@T failure criterion

Table 1 Mechanical data gaps associated with understanding Zircaloy performance after experiencing a transient beyond the critical heat flux

Item	Data Gaps	Materials	Material Property	Transient Temperature Conditions (°C)	Hold Time	Phase	Required Test Facilities	Notes
1	Unirradiated Heat Treatment Effects	Zircaloy-2 - RXA vs. SRA	Mechanical Properties	450–750	<1 to 15 s	1	t@T Transient: SATS or induction furnace	Zry-2 data is limited and specific to RXA and cold-worked. A comparison between RXA and SRA is necessary to assess whether their performance is similar or to identify any differences in performance.
2	Irradiation Temperature Effects	Zircaloy-2 – RXA vs. SRA and Advanced zirconium alloys – Nb-containing alloys and pRXA	Mechanical Properties	350–750	<1 to 15 s	2	Irradiation: HFIR or ATR t@T Transient: Hot cell SATS or induction furnace Hot cell mechanical test frames	Data does not exist for operating temperatures (300–400°C). Data is needed to verify the observed behavior.
3	Impact of Heat Treatment and Irradiation Effects	Advanced zirconium alloys – Nb-containing alloys and pRXA	Mechanical Properties	350–750	<1 to 15 s	1 and 2	Irradiation: HFIR and ATR t@T Transient: Hot cell SATS or induction furnace Hot cell mechanical test frames	No available data.
4	Yield Stress at Temperature	All Zircaloy materials	Mechanical Properties	350–750	N/A	1	t@T High-temperature mechanical test frames (non-rad)	Material yield strength at temperature could be the limiting criterion with modern alloys and

								advanced heat treatment techniques.
5	Impact of Alpha + Beta and Beta temperature regimes on post-transient mechanical performance	All Zircaloy materials	Mechanical Properties (embrittlement)	> 600–1000	<1s to 15 s	1	High-temperature mechanical test frames with high heating rate capability (non-rad)	This regime is specific to PWRs and does not apply to the re-use case.
6	Post-Transient Corrosion Kinetics	Advanced zirconium alloys	Corrosion	350–750	<1 to 15 s	1 and 3	t@T Transient: SATS or induction furnace (rad and non-rad) Unirradiated: Autoclave and MITR Re-irradiation: MITR and ATR	This regime is required for both PWR and BWR cladding materials. Work in this area intends to support the need for re-irradiation and long-term autoclave testing.
7	Transient creep down	All Zircaloy Materials	Creep	350–750	<1 to 15 s	3	t@T Transient: SATS Unirradiated: SATS Re-irradiated: ATR	Local geometry changes in the cladding may increase sensitivity to pellet–cladding interaction and pellet–cladding mechanical interaction type failures.
8	Post-transient operational creep performance	All Zircaloy materials	Thermal Creep	350–750	<1 to 15 s	1 and 3	t@T Transient: Hot cell SATS Re-irradiation: ATR	Changing the mechanical properties impacts creep behavior, but by how much remains unclear.

Table 2 Modeling and simulation gaps associated with understanding Zircaloy performance after experiencing a transient beyond the critical heat flux

Modeling and Simulation Gaps	Description	Phase
Static recovery in Zircaloys	The kinetics of static recovery for different grades of Zircaloy are not well understood under temperature ramps. NEAMS has developed static recovery models that need calibration against experimental data.	1, 2, 3
Modeling dynamic recrystallization	Preliminary studies in the NEAMS program suggest that both grain size effects and defect annihilation mediated by DRX could largely influence the post-t@T response. Currently, NEAMS does not have models for dynamic recovery. Efforts will have to be focused on capturing the kinetics of recovery and their impact on grain size and texture.	1 and 2
Phase transformation	Through support from the AFC program, efforts have begun to assess the best route to account for the effect of Alpha + Beta phases and transitions on the system's mechanical response. Yet, thus far, polycrystal models do not account for such effects. Extensions of the current state-of-the-art models will therefore be necessary.	1
Irradiation defect recovery kinetics	NEAMS is currently developing models tracking irradiation-induced defects. However, these models will need significant extensions to capture rapid dissolution due to t@T.	2 and 3

3. PROJECT TIMELINE

The project is organized into three phases, as illustrated in Figure 1. The President's Executive Orders emphasize the need for a large-scale expansion of nuclear power. Therefore, testing priorities will be focused on reactor types and cladding materials that offer the greatest impact for stakeholders and the United States. Each phase progressively deepens the investigation into the phenomena of interest, with the phases defined as follows.

- **Phase 1:** Testing on unirradiated and test reactor irradiated cladding
    - List of proposed milestones
        - Recrystallization testing on unirradiated cladding with different heat treatments
        - Corrosion post-irradiation examination (PIE) cladding after a temperature transient
        - Development and refinement of a high-temperature yield stress model for Zircaloy cladding
        - Creep down of Zircaloy tubes during an overpressure temperature transient
        - Impact of oxide or hydrogen embrittlement following high-temperature transients
        - Impact of heat treatment on irradiation recovery following a temperature transient
- **Phase 2:** Testing on test reactor and commercially irradiated cladding
    - List of proposed milestones
        - Impact of localized irradiation recovery on post-transient thermal creep behavior
        - Impact on yield stress following a temperature transient on commercially irradiated material
- **Phase 3:** Re-irradiation
    - List of proposed milestones
        - Long-term creep and corrosion PIE irradiated cladding following a temperature transient
        - Long-term creep and corrosion PIE following re-irradiation of commercially irradiated cladding following a temperature transient

Each phase will consist of various AFC milestone reports, and raw data or analyzed data will be transmitted throughout to industry stakeholders. A high-level summary of the project timeline and cost estimates by fiscal year is presented in **Error! Reference source not found.**. Detailed descriptions of the technical scope and budget for each phase are provided in the following sections.

Table 3 High-level summary of the project timeline and cost associated with each phase

	FY26	FY27	FY28	FY29	FY30
Phase 1	\$1.7M	\$1.7M			
Phase 2	\$1.6–2.3M	\$1.6–2.3M	\$1.6–2.3M		
Phase 3	\$1.2M	\$1.5M	\$2.25M	\$2.25M	\$1.5M
Total funding by year	\$5.2M	\$5.7M	\$5M	\$2.25M	\$1.5M

3.1 Phase 1 – FY26 to FY27

As-received cladding can be affected by thermal transients if they alter the material's original heat treatment (annealing state) from manufacturing. Such transients may occur early in the fuel cycle, with t@T conditions sufficient to modify the as-manufactured microstructure. For example, heating above the alpha-phase followed by rapid quenching can significantly influence material properties such as tensile strength, creep resistance, recrystallization behavior, and corrosion performance.

Changes in these properties that occur before significant irradiation-induced hardening can be limiting for certain cladding materials or heat treatments. SRA or pRXA alloys may be more susceptible to microstructural evolution than fully recrystallized alloys and may therefore exhibit reduced performance, particularly in creep and tensile strength, even in the unirradiated condition.

Phase 1 will focus on characterizing the following degradation mechanisms that could prevent the reuse of fuel rods or assemblies:

1. Recrystallization and irradiation recovery
2. Yield stress exceedance
3. Enhanced localized creep
4. Accelerated corrosion

The primary objective of Phase 1 is to collect data on unirradiated materials related to these phenomena and to develop plans for subsequent irradiation campaigns, particularly to study irradiation recovery. Accordingly, Phase 1 will target items 1, 3, 5, 7, 8, and 9 from Table 1. The data obtained will also be used to validate NEAMS models as outlined in

Table *2*.

t@T testing involves subjecting materials to rapid thermal transients (>50 °C/s) to simulate loss of heat transfer, such as that which would occur during a post-CHF event. These transients typically last seconds but may extend to several minutes for longer-duration events. While the thermal conditions vary between PWR and BWR scenarios, there is some overlap based on modeling predictions. Table 4 outlines the proposed test types and number of tests needed to address the selected items from Table 1. Simulating t@T transients may be achieved using induction, radiative, or other heating methods, provided the sample temperature is accurately controlled and characterized. Pre- and posttest microscopy will be performed to evaluate α-Zr grain growth, recrystallization extent of partially crystallized cladding, SPP density, and irradiation defects including a-loop and c-loop dislocation density. However, it is challenging to performed detailed characterization on Zircaloy materials, particularly with cold worked material due to the highly defected and fine grain regions, which imposes some uncertainty in the overall timeline. Lastly, model validation and subsequent development activities will be conducted in parallel and complementary to all testing.

Each test or material characterization is estimated to cost approximately $10k. In parallel, a dedicated modeling and simulation effort is required to support the experimental activities and guide test planning. Ideally, 1.5 full-time equivalent laboratory staff will be engaged in the t@T modeling work, contributing to an estimated annual cost of $600k.

Based on the planned number of tests, associated modeling support, and project duration, the total estimated cost for **Phase 1 is approximately $4.4 million**.

Table 4 Proposed number of tests for Zircaloy BWR and PWR materials

Mechanical test	Number of materials	Number of conditions per material	Total number of tests
Yield Stress at Temp.	2 PWR and 4 BWR	16	96

Post-transient tensile	2 PWR and 4 BWR	9	54
Post-transient thermal creep	2 PWR and 4 BWR	6	36
Post-transient In-Pile Corrosion	2 PWR and 4 BWR	6	36
Pre- and Post-transient Microstructure	2 PWR and 4 BWR	6	36

Table 5 Representative t@T transient to support each reactor type

Reactor Type	a/(a+b) Phase Estimate	Temperatures	Duration of conditioning (seconds)
BWR	860	450, 600, 750, and 800	1–3,10–20, 60–100, 600–5000
PWR	~710–740	700, 800, 900, and 1000	1–3,10–20, 60–100

3.2 Phase 2 – FY26 to FY28

Phase 2 is a natural extension of Phase 1, expanding the scope to include irradiated cladding materials. After irradiation, all zirconium alloys exhibit increased tensile strength, generally improved irradiation and thermal creep resistance, and reduced ductility. These changes are primarily due to radiation-induced increases in dislocation density and the formation of dislocation loops. However, the extent to which these irradiation-induced microstructural features influence engineering properties following thermal transients—particularly annealing effects during t@T events—is not well understood. Although prior testing of unirradiated cladding (e.g., NUREG-0562) has shown that t@T-induced effects can resemble those observed in irradiated material, these phenomena remain inadequately characterized in the irradiated state. Phase 2 aims to address key data gaps identified in Table 1—specifically items 2, 3, 4, 7, and 9.

Recent testing of modern recrystallized Zircaloy suggests that post-transient recrystallization may not occur, and thus the focus has shifted toward understanding irradiation annealing rather than recrystallization. While irradiation is not expected to significantly impact post-transient corrosion, confirmatory testing is warranted. Of particular concern is enhanced localized creep: because only small regions of the cladding may experience t@T transients, localized changes in creep behavior could compromise cladding integrity. Phase 2 will focus on characterizing the following degradation mechanisms that could inhibit the reuse of fuel rods or assemblies:

1. Irradiation recovery
2. Accelerated or abnormal corrosion (confirmatory)
3. Enhanced localized creep

In FY26, Phase 2 will prioritize the design, fabrication, and deployment of test reactor irradiation campaigns to achieve the required dose levels (i.e., equivalent fluence) for cladding specimens. Where feasible, commercially irradiated cladding materials will also be identified and harvested to supplement the testing program. Test conditions for irradiated material will be informed by Phase 1 results and implemented in FY28 and FY29. Further, pre- and posttest microscopy will be performed to evaluate α-Zr grain growth, recrystallization extent of partially crystallized cladding, SPP density, and irradiation defects including a-loop and c-loop dislocation density. However, it is challenging to performed detailed characterization on Zircaloy materials, particularly with cold worked material due to the highly defected and fine grain regions, which imposes some uncertainty in the overall timeline. Due to the significantly higher cost of testing irradiated materials, the number of tests must be limited. Table 6 outlines a representative test matrix designed to address the critical data gaps identified in Table 1.

Table 6 Proposed number of tests for Zircaloy material

Mechanical test	Number of materials	Number of conditions per material	Total number of tests
Post-transient tensile	2 – different heat treatments	6	12
Post-transient thermal creep	2 – different heat treatments	2	4
Post-transient Corrosion	2 – different heat treatments	2	4
Pre- and Post-transient Microstructure	2 – different heat treatments	Confirmatory	As needed

Irradiation build costs vary significantly across laboratories, making precise cost estimation difficult. For planning purposes, it is assumed that each irradiation capsule will be an off-the-shelf design, with an estimated cost of $300k to $500k per capsule. One capsule will be required for each material and test scenario outlined in Table 6, resulting in a total of eight capsules. In addition, each material test or characterization is estimated to cost approximately $30k to $50k. To support experimental planning and data interpretation, a dedicated modeling and simulation effort is essential. This will involve approximately 1.5 full-time equivalent (FTE) laboratory staff, contributing to an annual cost of roughly $600k.

Considering the number of capsules, associated material testing, modeling support, and the expected duration of Phase 2, the **total cost of Phase 2 is estimated to range from $5 million to $7 million**. The primary driver of cost variability is the irradiation builds. These costs may be reduced if commercially irradiated materials can be sourced and used in place of new test reactor irradiations.

3.3 Phase 3 – FY29 to FY33

Phase 3 is designed to resolve any outstanding issues arising from the NRC's Requests for Additional Information (RAIs). The primary reason is that the BWR I-loop at Idaho National Laboratory (INL) is not expected to come online until FY28; however, resources are still needed to continue establishing this capability. That said, this phase specifically targets issues that can only be addressed through re-irradiation and transient testing. As such, the scope includes transient testing (t@T) followed by subsequent re-irradiation. To optimize resources, the most efficient approach involves conducting t@T testing on defueled samples at the Oak Ridge National Laboratory (ORNL) Severe Accident Test Station (SATS). Following testing, the samples would be shipped to INL for refueling and refabrication into rodlets. These rodlets would then be irradiated in the Advanced Test Reactor (ATR) using the ATF Loop experimental capabilities. PIE would primarily focus on assessing post-transient corrosion behavior, including oxidation and hydrogen pickup. This strategy minimizes the need to develop new, costly, and time-intensive transient testing infrastructure, thereby accelerating data acquisition to support the LWR industry.

This effort is intended to serve a confirmatory purpose; therefore, testing should focus on two materials subjected to three distinct t@T transients. These transients will be selected based on insights from both unirradiated and irradiated test results.

Based on this assumption, the cost breakdown is as follows:

- A single test at the SATS is estimated at approximately $200k.
- Shipment and refabrication of each sample are projected to cost around $500k.
- The subsequent irradiation is expected to cost an additional $250k.

- PIE, limited to oxide thickness measurement and hydrogen pickup analysis, is estimated at $250k.

Given this structure, **Phase 3 is anticipated to require a total investment of approximately $7.2 million.**

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