U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

PROJECT PLAN TO PREPARE THE U.S. NUCLEAR REGULATORY COMMISSION FOR EFFICIENT AND EFFECTIVE LICENSING OF ACCIDENT TOLERANT FUELS

Version 1.1

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1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) is committed to enabling the safe use of new technologies, especially those that can increase the safety of NRC -regulated facilities. The U.S. nuclear industry, with the assistance of the U.S. Department of Energy (DOE), plans to deploy batch loads of some accident tolerant fuel (ATF) designs in the operating fleet on an aggressive timeline (by the early to mid-2020s). The NRC is optimistic that its preparation strategy and new paradigm of fuel licensing outlined in this project plan will support that schedule while still providing reasonable assurance of public health and safety at U.S. nuclear power plants. The NRC understands that it may face challenges in its preparations and technical and licensing reviews, but it is committed to working through such challenges in a thoughtful and deliberative manner.

In an attempt to increase regulatory stability and certainty along with enhancing and optimizing NRC review, the staff has developed this plan, which includes a vision for a new paradigm for ATF licensing. The staff believes that adherence to this strategy will benefit all the agency's stakeholders in the planned deployment of ATF designs.

The NRC staff has extensively engaged with its stakeholders in the development and finalization of this project plan, consistent with the NRC's principles of good regulation and statutory requirements. The staff has held numerous public meetings with external stakeholders, including licensees, nuclear fuel vendors, industry groups, nongovernmental organizations, and international counterparts. The staff has found these interactions invaluable and has considered the views and comments of the NRC's stakeholders in finalizing this version of the ATF Project Plan. A separate, companion document to this plan compiles the staff's resolution of the feedback received during the official public comment period and is available at Agencywide Documents Access and Management (ADAMS) Accession No. ML18261A415.

The project plan presents the high -level strategy that the staff will follow to ensure that it is ready to receive ATF topical reports (TRs) or licensing actions for review. At this point, the strategy is concept and technology independent. Concepts are defined as a family of ATF designs with largely similar characteristics. Examples include chromium (Cr)-coated zirconium (Zr) alloy claddings, steel claddings, silicon carbide (SiC) claddings, or metallic fuels. Individual vendors may implement variations within each concept as specific technologies. For example, the different methods that are used to apply Cr coatings would be identified as separate technologies.

2 BACKGROUND

In a coordinated effort under the direction of the NRC's ATF steering committee, the Office of Nuclear Reactor Regulation (NRR), Office of New Reactors (NRO), Office of Nuclear Material Safety and Safeguards (NMSS), and Office of Nuclear Regulatory Research (RES) are preparing for the licensing and use of ATF in U.S. commercial power reactors.

ATF Steering Committee		
Office of Nuclear Reactor Regulation	**Office of Nuclear Regulatory Research**	**Office of Nuclear Material Safety and Safeguards**
• **Division of Safety Systems *(Chair)*** • **Division of Risk Assessment** • **Division of Operating Reactor Licensing**	• Division of Systems Analysis • Division of Risk Assessment	• Division of Materials Safety, Security, State, and Tribal Programs • Division of Fuel Management

Figure 2.1 The NRC's ATF steering committee

In coordination with DOE, several fuel vendors have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (i.e., fuels with longer coping times during loss of cooling conditions). The designs considered in the development of this plan, both within and outside of the DOE program, include coated Zr claddings, doped uranium dioxide (UO_2) pellets, iron -chrome -aluminum -based (FeCrAl) cladding, SiC cladding, uranium silicide (U_3Si_2) pellets, and metallic fuels (e.g., Lightbridge). For these ATF designs, the industry's stated schedules for the initial irradiation of lead test assemblies (LTAs) and the review of TR and license amendment requests (LARs) were used as a basis for the timelines discussed in this plan.

This project plan covers the complete fuel cycle, including consideration for the front and back ends (e.g., fabrication, transportation, and storage), and outlines the strategy for preparing the NRC to license ATF designs with a focus on the preparation, review, and approval of TRs. It also identifies the lead organization for each planned activity. The plan only briefly touches on existing licensing activities, such as the TR process, the implementation of LTA programs, and the LAR process, as such activities follow existing processes that have well -established schedules and regulatory approaches or are being clarified through NRC initiatives outside of the ATF steering committee and working group.

In preparing the agency to conduct meaningful and timely reviews of these advanced fuel designs, the NRC is reviewing the existing regulatory infrastructure and identifying needs for additional analysis capabilities and developing unique critical skillsets within the staff. The NRC has entered a memorandum of understanding (MOU) with DOE to coordinate on the nuclear

safety research of ATFs that will make the appropriate data available for regulatory decision -making processes. In addition, the NRC has established an MOU with the Electric Power Research Institute (EPRI) to facilitate data sharing and coordination on expert elicitation.

For the purpose of developing this plan, ATF concepts are broadly categorized as near term and longer term. The plan considers near -term ATF concepts as those for which the agency can largely rely on existing data, models, and methods for its safety evaluations (SEs). Coated Zr cladding, FeCrAl cladding, and doped UO_2 pellets are a few examples of near term ATF concepts. In general, the industry is pursuing these near term concepts for deployment by the early to mid2020s. Longer term ATF concepts are those for which substantial new data, models, and methods need to be acquired or developed to support the agency's SEs. U_3Si_2 fuel, metallic fuel, and SiC ----based cladding are a few examples of longer term ATF concepts. Note that "near term" and "longer term" are terms of convenience used to indicate the current expected deployment timeframe for the ATF concept.

Regulatory requirements do not vary between near -term and longer -term concepts, and the NRC will evaluate all designs based on their individual technical basis. The timeline for licensing will be commensurate with the deviation of the ATF technology from the current state of practice and the number and complexity of issues related to phenomena identified during an expert elicitation process (e.g., a phenomena identification and ranking table (PIRT)).
The agency is focusing its current ATF licensing preparation on the use of ATF in light -water reactors (LWRs) in the operating fleet. Some overlap may occur between LWR ATF fuel development and fuel safety qualification of some types of non -LWR fuels for advanced reactor designs. As appropriate, the NRC will leverage previous experience to help optimize licensing efficiency and effectiveness.

This project plan will be a living document that may evolve as (1) ATF concepts are more clearly defined, (2) schedules are refined, (3) the knowledge level of specific concepts increases as experimental testing programs are completed, and (4) potential extensions to the current operating envelope of fuel are identified.

3 ACCIDENT TOLERANT FUEL LICENSING PROCESS

This project plan focuses on the NRC's preparations to conduct efficient and effective reviews of TRs for ATF designs on a schedule consistent with published industry timelines. TRs provide the generic safety basis for a fuel design and do not, by themselves, grant approval for operating plants to begin loading ATF. These reviews for new fuel designs have historically taken between 2 and 3 years to complete. Based on past experience, vendors should also anticipate that the NRC's Advisory Committee on Reactor Safeguards may request to review ATF TRs and should include time for such reviews in their planning and schedules.
In addition, a licensee may need to submit an LAR to modify its license to allow for the use of a new fuel design. LARs address all plant -specific aspects of implementing the ATF concept and are typically completed on a 1-year review schedule.

Upon final approval of the plant -specific LAR, a licensee would be authorized to load and irradiate batch quantities of the specific ATF design in accordance with its license.

3.1 Milestone Schedule

Table 3.1 outlines some of the high -level milestones and associated dates related to implementation of the ATF project plan. These dates are based on the staff's current state of knowledge at the time of publishing.

Table 3.1 ATF Milestone Schedule

Milestone/Activity	Schedule
Complete ATF project plan V1.0.	September 2018 [C]
Issue final LTA guidance.	June 2019 [C]
Conduct PIRT for coated claddings.	July 2019 [C]
Revise ATF project plan (V1.1) to include the fuel burnup and enrichment extension preparation strategy addendum	November 2019 [C]
Complete supplemental guidance regarding the chromium-coated zirconium alloy fuel cladding ATF concept	Early 2020
Identify and implement adjustments to the regulatory infrastructure, if necessary, to enable the full potential of ATF (e.g., increase enrichment, increase burnup).	2019–2025
Conduct TR reviews for near -term concepts.	2020–2022
Conduct LAR reviews for near -term concepts.	2022–2023
Conduct TR/LAR reviews for longer term concepts.	TBD (in accordance with industry schedules)

[C] denotes completion of activity.

3.2 Assumptions

Given the current uncertainty related to the development and deployment of ATF designs, the NRC staff made the following major assumptions to help in its development of this plan:

- The NRC will not need to perform independent confirmatory testing for specific ATF designs. The NRC expects that the applicant, DOE, or other organizations will provide the agency with all data needed to support the safety basis for a concept. Additionally, the NRC expects that all reactor and test -generated fuel behavior data will be provided to the agency in a timely manner so that it can assess NRC analysis capabilities.
- Interaction with DOE, EPRI, vendors, and other organizations involved in ATF -related experimental programs will take place in real-time and, whenever possible, in advance of experiments being conducted.
- The NRC's interactions with external stakeholders will keep the staff and stakeholders informed about both technical and programmatic developments that are affecting activities identified in this project plan.

3.3 New Fuel Technical Basis Development Process

The development of the technical basis necessary to qualify a new fuel design is an extensive process, both in terms of resources and time. Figure 3.1 depicts the basic steps applicants take toward obtaining the appropriate data and experience needed to license a new fuel design for batch loading (typically one -third of the fuel assemblies in the reactor core per cycle) at a U.S. commercial nuclear power plant.

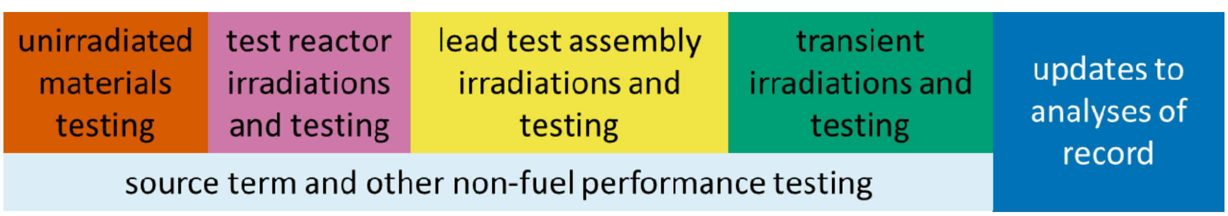

Figure 3.1 New fuel design technical basis development process

Figure 3.1 depicts the relative order of testing needed to develop the technical basis for a new fuel design, including ATF. The first box in orange (unirradiated materials testing) represents the testing necessary to characterize the material, mechanical, chemical, and thermal properties of a new design. The second box in pink (test reactor irradiations and testing) represents the vendor's characterization of the evolution of those properties obtained in the first box with irradiation and time spent in the reactor. The third box in yellow (LTA irradiations and testing) provides the integral testing to fully characterize the fuel in prototypical operating conditions and donor material for the next step. The fourth box in green (transient irradiations and testing) is focused on the use of fuel segments harvested from LTAs to perform tests that mimic transient and accident conditions. Such tests are key to ensuring safety. The fifth box in light blue (source term and other non-fuel performance testing) happens concurrently with these four

tasks. It includes testing to characterize fission product release, core melt progression, core relocation, and mechanical and chemical interactions. Finally, the sixth box in dark blue (updates to analyses of record) involves the development, calibration, verification, and validation of analytical models to simulate the performance of the new fuel design under normal and accident conditions. This also requires the quantification of uncertainties and the definition of an application methodology.

Even under ideal circumstances, the time and effort required to fully develop the technical basis are substantial largely because of the irradiated testing needed to fully understand and characterize how a design or material acts under steady -state, transient, and accident conditions. Understanding these characteristics and being able to model them appropriately represents the critical path to the licensing of ATF.

Advancements in instrumentation and examination equipment used to collect data from irradiated fuel may allow for information to be gathered either more quickly following irradiation or at a higher fidelity than in the past. Progress in this area could expedite the timeline required to document the experimental and testing data required to fully support the technical basis for an ATF concept. Staff will maintain engagement with industry progress in this area to ensure preparation for new or novel applications.

The NRC understands that vendors and organizations participating in the development of the technical basis for ATF designs could possibly leverage advanced modeling and simulation capabilities to expedite the development timeline. Insights gained from mechanistic computational tools could improve testing programs and thus limit or eliminate failed experiments. This could allow for data that support the technical basis and ultimately an TR to be gathered more quickly.

The staff is currently aware of the latest developments in modeling and simulation credibility, as reflected in the literature and in standards on the topic, which echo the need to perform experimental testing. The staff does recognize the need to maintain engagement and continues to monitor state-of-the----art advances that could potentially contribute to substantial timeline constrictions, especially for longer term concepts.

3.4 Project Plan Paradigm

This project plan envisions an improved fuel licensing paradigm, depicted in Figure 3.2, that can increase the efficiency and effectiveness of the NRC's review of ATF designs.

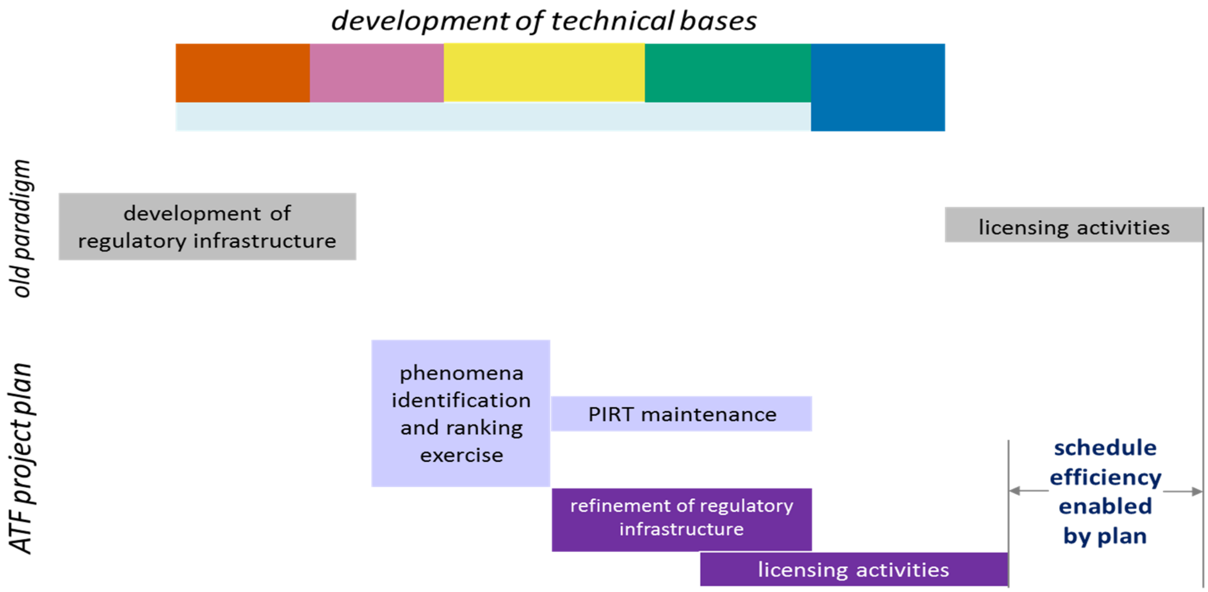

Figure 3.2 ATF project plan new paradigm

3.4.1 Old Paradigm

In the old paradigm, the regulatory infrastructure was essentially developed before, or soon after, commencement of technical basis development activities, such as experimental testing programs, because, over the past several decades, most LWR "new" fuel designs and characteristics have been relatively minor changes within the UO_2-Zr alloy fuel matrix. Licensing activities would then commence upon completion of the development of the technical basis, with little or no interaction with the NRC staff along the way. This lack of interaction has produced several issues, including cases in which additional testing that required irradiation was necessary before NRC approval, which caused extensive delays.

3.4.2 New Paradigm

Recent initiatives related to ATF are leading to the largest potential departures from the basic UO_2-Zr alloy fuel matrix, which has been used in U.S. and foreign reactors over the past 50 years. This potentially transformational technology that industry is pursuing has led the staff to reflect on the NRC's fuel licensing process and determine where improvements to efficiency and effectiveness can be gained. The goal of this new paradigm is to enhance regulatory stability and add efficiency to the timeline required for licensing activities following the completion of the technical basis to support an ATF design.

As illustrated in Figure 3.2, the schedule efficiency enabled by the new paradigm requires the staff to conduct thorough and meaningful PIRT exercises for each ATF concept and maintain the results of the PIRT as the collective state of knowledge evolves. The outcome of the ATF PIRT process, elaborated on below, will allow the staff to refine the regulatory infrastructure, as needed, for each concept and facilitate the development of concept -specific licensing roadmaps. These activities will proceed in parallel with the continued development of the full technical basis by the vendor.

Refinement of the regulatory infrastructure will be done in real-time and with significant communication with agency stakeholders to maintain transparency and clearly communicate regulatory expectations to the vendors as early as possible in the process. In addition, the new paradigm allows for licensing activities in the form of TRs to proceed in parallel with the completion of the technical basis for a specific concept. Data sharing and close engagement with the vendor during this time will be critical in gaining the efficiency depicted in Figure 3.2. The staff will need to perform significant amounts of prework to prepare for and conduct the most efficient reviews of ATF LTRs.

The overall goal of this strategy is to develop and communicate the NRC's expectations for the technical basis of specific ATF concepts in real-time to minimize the lag between the completion of the technical basis and the licensing of ATF designs to their full potential.

3.4.3 Phenomena Identification and Ranking Table Exercises

As stated above, the success of the strategy outlined in the project plan requires the staff to conduct thorough and meaningful PIRT exercises for each concept and maintain the results of the PIRT as the collective state of knowledge for each concept is advanced. For the purpose of this project plan, the term PIRT is defined as an expert elicitation process in which panelists will identify and rank new phenomena important to safety introduced by an ATF concept. The staff imagines that these exercises will vary greatly in scope and depth based on the departure of the concept from the current state of practice and the maturity of the concept. Some examples of potential exercises include independent, NRC review of an industry generated failure -mode analysis, a coordinated NRC and vendor exercise on a vendor specific concept, and a multi-day PIRT panel with topical experts similar to previous NRC PIRTs such as on high -temperature gas reactors.

The experts selected for the PIRT panel should consider the full intended use of the concept to ensure that the PIRT results are meaningful even if initial licensing applications do not intend to seek credit for the enhanced capabilities of the concept. Lack of consideration of the full operating envelope in the initial PIRT exercise could lead to uncertainty further along in the process when a vendor or licensee does seek to credit those capabilities.

The NRC staff relies on the agency's significant expertise in the Zr-clad UO_2 fuel system during the review of current fuel licensing submittals. However, the staff does not necessarily have this

same level of knowledge for all the ATF concepts that industry is currently pursuing. The NRC staff is monitoring the literature and experimental testing programs conducted in the public domain and is participating in industry and DOE update meetings on ATF concept development. However, more in -depth expertise may be needed to support the efficient and effective review of ATF licensing submittals. PIRT exercises will allow the staff to benefit from external expertise to identify phenomenon important to safety for each concept and, therefore, to refine the regulatory framework that is necessary for a concept ahead of licensing submittals and that will serve as baseline guidance for the NRC's technical review.

In addition to concept -specific PIRTs, discipline -specific PIRTs may be useful in some cases. Examples considered to date include PIRTs in the areas of severe accidents, storage and transportation, burnup above 65 gigawatt -days per metric ton of uranium and enrichment above 5 weight percent. The experts necessary to identify and evaluate new phenomena important to safety in these areas should be the same or similar experts for all or many of the ATF concepts under development. Therefore, the NRC staff believes that it would be more efficient to conduct these PIRTs in a discipline -specific manner instead of as part of the concept -specific exercises.
For each concept, a portion of the information needed to examine the phenomena important to safety is generic to the concept and publicly available, whereas a portion of the information is proprietary and specific to a fuel vendor's technology. Figure 3.3 illustrates this relationship.

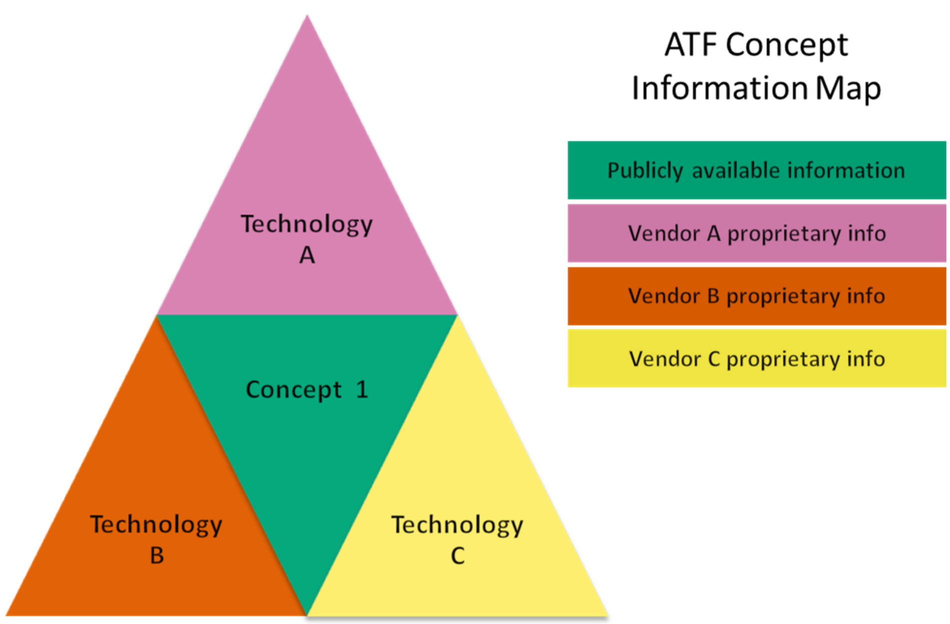

Figure 3.3 ATF concept information map

For concept -specific PIRTs, the series of events will generally follow the sequence depicted in Figure 3.4, whereby an expert panel collects, synthesizes, and reviews publicly available information pertinent to the concept. In this model, information specific to a vendor's technology is reviewed separately after the results of the PIRT exercise have been documented and as far ahead of the submittal of TRs as possible.

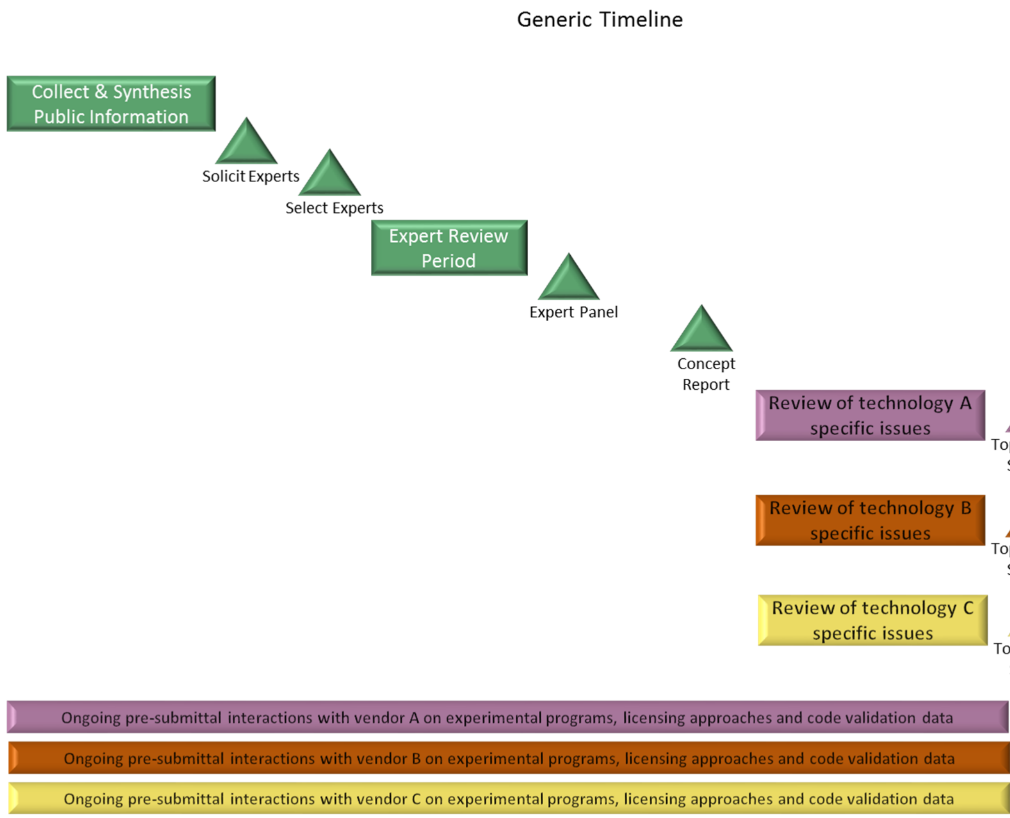

Figure 3.4 Generic ATF PIRT exercise timeline

The NRC completed the first ATF PIRT exercise on Cr coated- cladding in June 2019. The PIRT began by collecting publicly available data on coated cladding concepts and producing an initial report, which was completed in January 2019. This report was used as background material for the experts who participated in the panel discussion and provided input to the final report. This followed the schedule in Version 1.0 of the project plan, as depicted in Figure 3.5.

Experts participating in the panel had background from academia, national labs, the nuclear industry and high temperature coatings. A multi-day public meeting was held where the experts discussed the initial report and their areas of expertise. After rating a list of fuel damage mechanisms by importance and level of knowledge the report was finalized.

This final PIRT report was then used to inform the development of interim staff guidance (ISG) on coated cladding. This guidance will be used to inform NRC staff reviews of coated cladding topical reports and license amendments and will ultimately be incorporated into the Standard Review Plan (SRP).

While the ISG is being produced on an expedited timeline to facilitate issuance prior to the anticipated topical report submittals on coated cladding, the NRC staff have made efforts to include stakeholders in the process. This effort has included opening the PIRT up as a public meeting, holding multiple public meetings on the ISG, and noticing the ISG in the federal

register for public comment. The current expected date for final issuance is around February of 2020.

The NRC will develop timelines for subsequent ATF PIRT exercises and additional implementation details through coordination with its external stakeholders.

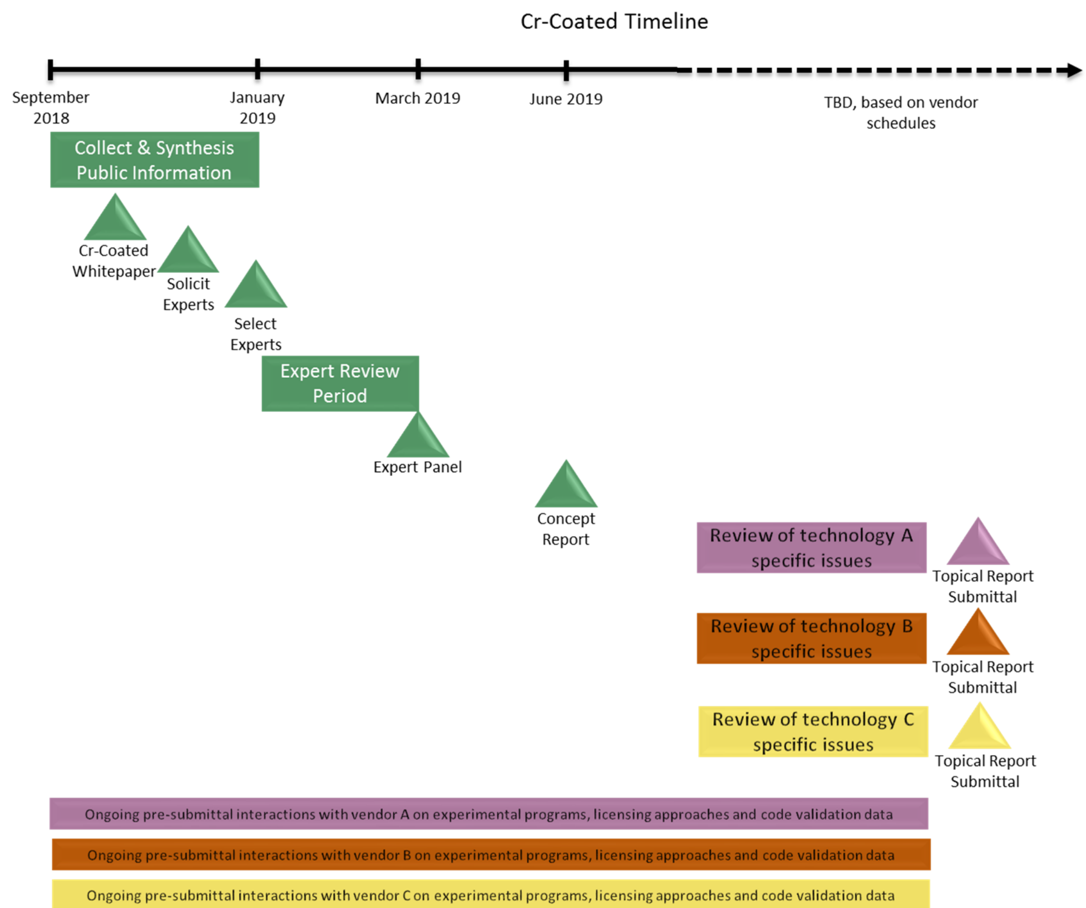

Figure 3.5 Cr-coated cladding PIRT exercise timeline

3.4.4 Concept -Specific Licensing Roadmaps

Given that this project plan is high level and concept independent, it will be augmented by concept -specific licensing roadmaps. The staff will use the outcome of the expert elicitation process completed for each ATF concept and the strategy outlined in this plan to create these roadmaps, which will be made publicly available as addenda to this plan. The roadmap for each concept will outline a pathway for vendors to ensure that the technical basis, which they will submit to the NRC in a TR, will meet the staff's expectations. These roadmaps will identify any gaps or deficiencies in the regulatory framework for an ATF concept and will clearly note where the current framework applies and is sufficient. The NRC will develop the roadmaps in a transparent manner with opportunities for stakeholder feedback before their finalization.

The vendor will then take these roadmaps and develop, or modify, their strategy for licensing each specific ATF technology. Figure 3.6 shows this concept.

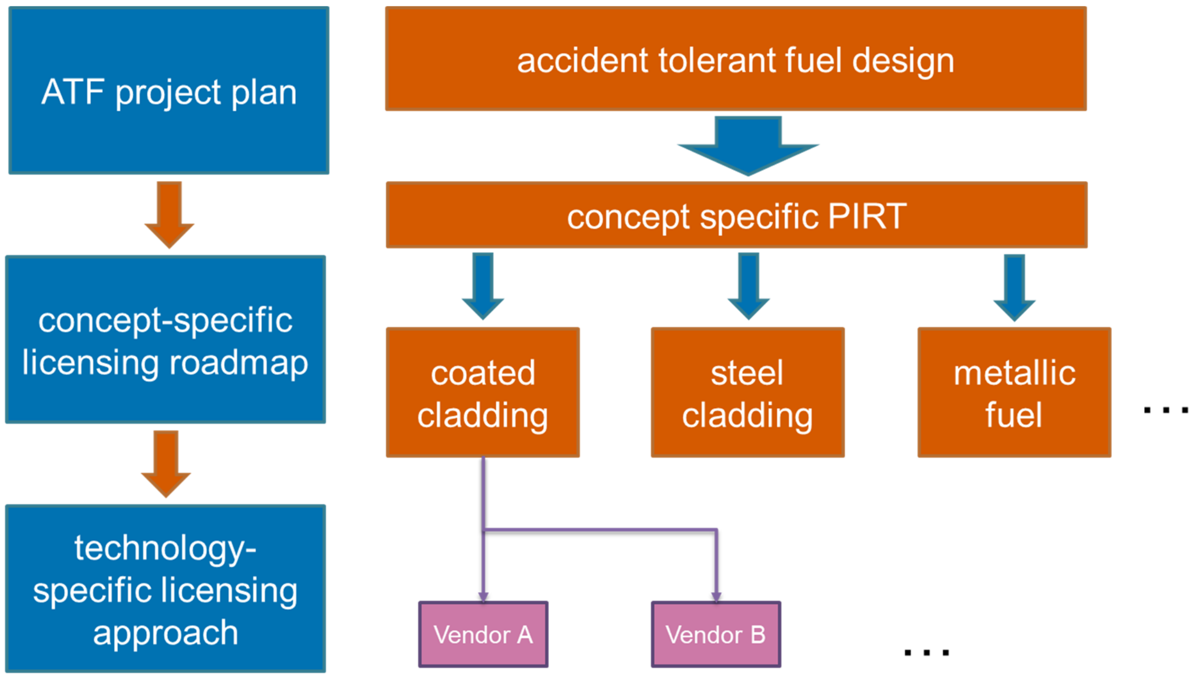

Figure 3.6 ATF concept -specific licensing roadmap

The staff will assess the results of the concept -specific expert elicitation process against the current licensing and regulatory framework (e.g., Title 10 of the *Code of Federal Regulations* (10 CFR); NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP); regulatory guides (RGs); and NUREG reports) to determine which regulations and guidance remain applicable and to identify (1) the areas for which an applicant would need to propose new acceptance criteria or (2) an alternative means for meeting the regulations. The NRC will develop these roadmaps within 6 to 12 months of completion of the concept -specific PIRT.

The staff is currently assessing the best vehicle for documentation of these roadmaps. Potential options included addenda to this plan, interim staff guidance, endorsement of industry developed guidance, or a safety review of vendor generated fuel qualification plans. The appropriate vehicle is likely to vary based on the concept and the level of departure from current technology and how many vendors are pursuing the concept. It is the goal of the staff to use the most efficient tool for each individual situation and not to mandate a one-size-fits-all solution.

The staff plans to complete these roadmaps sufficiently ahead of TR submittals so that vendors are able to ensure that their planned experimental testing programs will adequately address the NRC’s needs for approving the design.

3.4.5 Additional Efficiencies

The staff is exploring additional innovative and transformative ideas and solutions to further improve the agency's approach. A few examples that the NRC is currently pursuing include the following:

- development of a standard TR change process
- use of DOE and advanced computational capabilities
- expedited issuance of NRC guidance
- leveraging best practices from research and test reactor fuel qualification plans

In addition, the staff welcomes any additional suggestions from stakeholders of other potential efficiencies that could aid NRC in increasing the efficiency and effectiveness of its preparation and reviews of ATF designs.

3.5 NRC Approach to Confirmatory Analysis

The staff's approach to independent confirmatory analysis of TRs and LARs follows a graded approach. This approach varies based on the complexity of the application, the safety significance of the issues presented, and the uncertainty of the key phenomena involved.

The NRC uses a range of tools to verify the safety case made and presented by an applicant. In some instances, the staff can perform its confirmatory analysis and reach a safety determination by drawing on previous knowledge, accumulated expertise, and the information presented by the applicant. In other cases, confirmatory calculations performed by the staff can allow for a more effective and efficient review. The staff typically performs independent confirmatory calculations to assist in reaching a safety finding in reviews where the uncertainties are large, or the margin is small. In some cases of large safety significance and large uncertainty, the NRC has pursued independent confirmatory testing before reaching a determination on an application. Ultimately, the staff bases its safety finding on the technical basis and safety case provided by the applicant. Confirmatory analyses performed by the staff provide increased confidence in the applicant's results.

The NRC's approach for most ATF designs will likely follow the middle approach of performing independent confirmatory calculations to assist the staff in reaching a safety determination. In support of this, the staff is actively enhancing the NRC's suite of computational analysis tools and engaging DOE to understand the capabilities of its codes and methods.

4 STAKEHOLDER INTERACTIONS

Table 4.1 outlines key meetings and interactions scheduled during the development and review of ATF designs. The primary risks to timely licensing of ATF relate to current uncertainties in the schedules for necessary experimental programs. The staff intends to remain closely engaged with the organizations and entities acquiring data and adjust the plan as new information becomes available. The staff is closely following ongoing efforts to identify alternatives for testing that the shutdown of the Halden Research Reactor in Norway will affect. Another potentially significant risk to the successful implementation of ATF is a delayed recognition that changes to the regulations or regulatory guidance are required. The staff has initiated dialogue with stakeholders to communicate timelines required for various modifications to the regulatory infrastructure and to solicit input for changes that may be necessary for the different ATF concepts being explored.

4.1 Meetings, Stakeholder Interactions, and Critical Skill Development

The NRC is committed to engaging in industry project update meetings and supporting staff participation in experimental program discussions to maintain awareness of industry and DOE efforts and to prepare for regulatory reviews. The NRC will develop staff and contractors with critical skills required to support projected applications of high to moderate certainty. All stakeholder interactions will occur in accordance with the NRC's public meeting policy.

Table 4.1 Meetings and Stakeholder Interactions

Meeting	Frequency	Desired Outcome
EPRI/DOE/Idaho National Laboratory (INL) update meetings	Biannually	Assess the technical progress of ATF research and development (R&D). Obtain information necessary for developing analytical capabilities and licensing strategies.
TOPFUEL (rotates between the United States, Europe, and Asia)	Annually	Assess the technical progress of ATF R&D. Obtain information necessary for developing analytical capabilities and licensing strategies.
ATF standards and guidance development activities with the Organization for Economic Co-operation and Development /Nuclear Energy Agency, International Atomic Energy Agency, and international counterparts	Annually	Discuss licensing approach with international counterparts.
Fuel vendor update meetings (rotates from NRC Headquarters to the vendor's headquarters)	Annually (per vendor)	Assess the technical progress of ATF R&D. Obtain information necessary for developing analytical capabilities and licensing strategies (in addition to a number of other non-ATF outcomes).
ATR/TREAT test planning and test observation meetings	Annually	Develop an understanding of testing that will characterize the performance characteristics of ATF designs.
ATF fuel fabrication facilities tour and audit	As needed	Develop an understanding of manufacturing processes and obtain information for developing licensing strategies.

5 INITIATING STAFF ACTIVITIES

Because of design -specific aspects and schedules, the NRC's activities are linked to the industry's progress and plans to deploy ATF. For this reason, the agency must have a mechanism for communicating schedules and resource needs in advance of licensing activities. The staff is currently exploring the issuance of a generic communication (i.e., a regulatory issue summary (RIS)) to solicit information from industry on schedules and plans for ATF designs. The NRC expects the ATF RIS to be very similar to previously issued agency RISs on advanced reactors (ADAMS Accession No. ML17262B022) and molybdenum-99 production facilities (ADAMS Accession No. ML13078A385).

This plan provides estimated lead times for each activity associated with preparing the agency to conduct an effective and efficient licensing reviews of ATF LTRs. As the NRC staff gains more experience with these reviews, it will adjust lead times to account for difficulties or efficiencies, as necessary. These lead times dictate the amount of time ahead of licensing

activities that data should be provided and a formal communication of intent should be made through a response to the RIS, pre submittal meetings, or other formal interaction with the staff.

6 PREPARATORY ACTIVITIES

The staff has grouped its preparatory activities into four tasks. The highlights of each task are briefly described below; subsequent sections describe these tasks in full detail.

6.1 Task 1: Regulatory Framework, In-Reactor Performance

- Participate in coordinated PIRT exercises on in-reactor degradation mechanisms and failure modes under a wide array of accident conditions, performance -based metrics, and analytical criteria to ensure acceptable performance.
- Perform a scoping study to (1) evaluate the applicability of existing regulations and guidance for each ATF design, (2) identify changes to, or the need for, new regulations and guidance, and (3) identify any key policy issues.
- Identify consensus standards that need to be updated for ATF and participate in the update process where appropriate.
- Determine and clarify the regulatory criteria that need to be satisfied for partial or full core use of ATF and the regulatory options available to applicants and vendors.
- If needed, resolve policy issues and initiate rulemaking and guidance development activities.

6.2 Task 2: Fuel Cycle, Transportation, and Storage Regulatory Framework

- 10 CFR Part 70, "Domestic Licensing of Special Nuclear Materials"; 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"; and 10 CFR Part 72, "Licensing Requirements for the independent Storage of Spent Nuclear Fuel and High -Level Radioactive Waste, and Reactor -Related Greater Than Class C Waste," are largely performance based; therefore, the staff does not anticipate identification of gaps or deficiencies in these regulations.
- Gaps in the review guidance may develop as the fuel cycle industry develops plans for manufacturing, transporting, and storing ATF. The NRC will monitor the fuel cycle industry's plans and identify and develop any necessary regulatory guidance in a timely manner.
- The NRC has not identified any fuel cycle facility licensing activities for near -term ATF concepts.
- The NRC has identified two reviews of LTA transportation casks for near -term ATF concepts.

6.3 Task 3: Probabilistic Risk -Assessment Activities

- The staff will evaluate how industry batch loading of ATF may affect the current risk -informed programs such as risk -informed technical specification initiatives 4b and 5b.
- The NRC's risk -informed oversight activities (e.g., the significance determination process) depend on standardized plan analysis risk (SPAR) models, which may need to be updated to reflect the batch loading of ATF.

6.4 Task 4: Developing Independent Confirmatory Calculation Capabilities

The NRC's independent confirmatory analysis of TRs and LARs follows a graded approach, which varies based on the complexity of the application, the safety significance of the issues presented, and the uncertainty of the key phenomena involved. Further, independent confirmatory analysis does not always require independent confirmatory calculations using NRC -developed tools. In some cases, the staff can perform its confirmatory analysis and reach a safety determination by drawing on previous knowledge, accumulated expertise, and the information presented by the applicant. For many incremental changes in fuel design, independent confirmatory calculations using NRC -developed tools were not necessary. The NRC typically performs independent confirmatory calculations to review cases in which adequate margin is essential to safety.

For initial ATF licensing for which limited data will be available to formulate and validate models, independent confirmatory calculations will likely be needed. In these instances, the staff that performs the confirmatory calculations must have a clear understanding of the assumptions and limitations of the analytical tools that it uses. The staff has experience using both NRC -developed and non -NRC codes for confirmatory calculations. In either case, to use a code for confirmatory calculations, the staff must understand the range of conditions for which the code has been validated and understand the nature of the validation database.

The staff will take the most efficient and effective approach to verify the safety case presented for each ATF concept. When confirmatory calculations are warranted, the staff will select an approach that weighs several factors. These factors include the level of effort necessary to modify and validate existing NRC codes and the level of effort needed to understand and validate a non -NRC code. For example, for ATF concepts that use coatings on a Zr -based alloy, little effort is required to modify existing codes to model coating performance, whereas a significant effort is required to adopt a new code. In addition, the selected approach will consider the review schedule and the required models for each ATF concept. The concept -specific licensing roadmaps will elaborate on the selected approach for each ATF concept.

Based on the information available to date, the staff believes it will be more efficient and effective to pursue relatively minor modifications to existing NRC codes to model near -term ATF fuel concepts. The NRC has specifically tailored its codes to evaluate regulatory

requirements and phenomena important to safety and has extensively validated them. These features make them easy for the staff to use and give the staff high confidence in the results that they provide. At this time, the NRC plans to modify those codes that are developed to analyze fuel performance, thermal hydraulics, neutronics, and severe accidents and source terms. In addition, the agency is considering modifying existing NRC codes to model more long -term ATF fuel concepts in cases that require minimal effort.

Where possible, the NRC will coordinate with DOE to reduce duplication of effort in accordance with the DOE-NRC MOU.[1]

6.4.1 Advanced Modeling and Simulation

NRC staff maintain an awareness of the advancements in modeling and simulation for nuclear applications. Staff have participated in training sessions for a number of DOE's advanced modeling and simulation tools and is continuing to learn about the capabilities of these tools. A recent targeted effort to couple the NRC's TRACE thermal -hydraulic code with DOE's BISON fuel performance code through MOOSE, an independent solver, has given the staff a significantly greater understanding of DOE's codes. The staff has also successfully combined elements of DOE's codes developed under the Consortium for the Advanced Simulation of Light -Water Reactors program into the NRC's neutronics code, SCALE. These efforts have demonstrated that specific opportunities exist to leverage elements of DOE's codes to improve the NRC's analysis capabilities. The staff expects to continue to follow DOE's development efforts in the area of advanced modeling and simulation and to search for opportunities to leverage their capabilities.

The staff is aware of efforts to use advanced modeling and simulation in a variety of applications or families of codes: mechanistic codes, steady -state codes, and transient codes. Although advanced modeling and simulation in mechanistic codes can inform experimental programs, improve upon highly empirical correlations, and identify testing priorities, current advanced modeling and simulation tools do not appear to be mature enough to substitute modeling for experiments because of the complex nature of fuel and reactor behavior. Further, the state of knowledge in many areas still only permits semi-empirical modeling of key phenomena. Validation of these tools against relevant data will be essential to demonstrate their potential to support licensing activities. The staff will continue to coordinate with DOE and the national laboratories to better understand the capabilities of the DOE codes to potentially reduce the number of time -consuming and costly experiments and demonstrations.

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1 The 2017 ATF addendum to the NRC-DOE MOU appears in the NRC Library under "Document Collections" and "Memorandum of Understanding" at https://www.nrc.gov/reading-rm/doc-collections/memo-understanding/2017/.

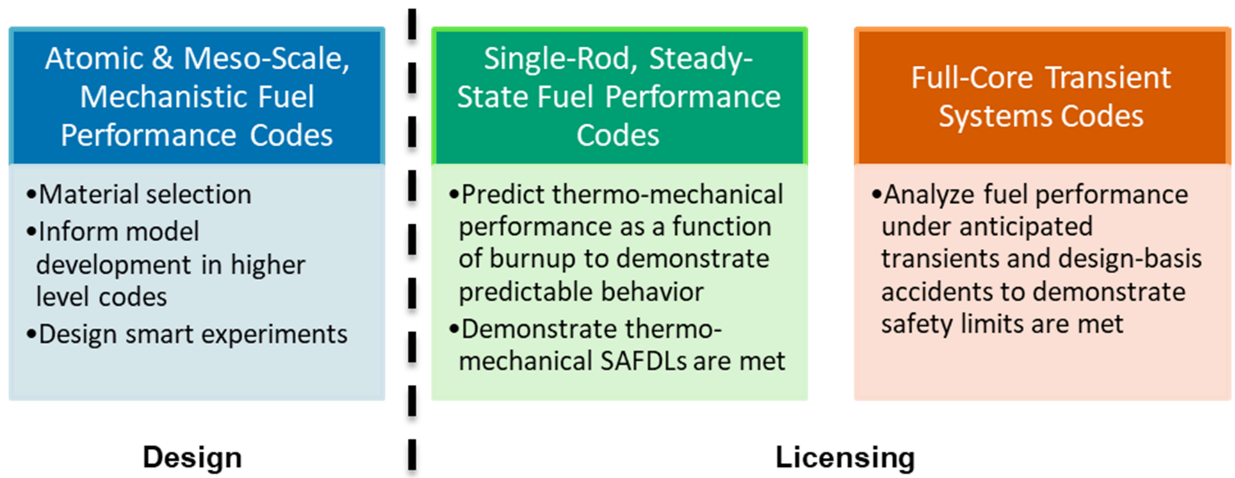

Figure 6.1 Example applications and use of code families in the area of fuel performance

At this time, the staff has had no indication from fuel vendors that they intend to rely on advanced modeling and simulation with atomic -scale, mechanistic modeling to support license applications for near -term concepts. However, these vendors have indicated interest in using these tools to design better experiments, inform model development in higher level codes used for licensing, and to supplement licensing applications for longer -term ATF concepts.

7 TASK 1: 10 CFR PART 50, 10 CFR PART 52, AND 10 CFR PART 100 REGULATORY FRAMEWORK, IN-REACTOR PERFORMANCE

ATF presents new and unique technical issues that current guidance, review plans, and regulatory criteria for UO_2-Zr-based nuclear fuel may not readily address. To prepare the agency to conduct meaningful and timely licensing reviews of ATF designs, well -developed and vetted positions on potential policy issues that may arise during ATF licensing are needed. These positions must be communicated to stakeholders clearly and early.

This plan contemplates two distinct ATF activities that may require changes to the regulatory framework: (1) batch loading of ATF into NRC regulated power plants and (2) crediting the safety enhancements of ATF in the licensing basis of NRC regulated power plants. The regulatory framework changes that may be necessary for each of these activities are likely to be different, and the staff anticipates that such changes will need to be made to address batch loading before making changes needed to credit the safety enhancements of ATF in the licensing basis.

The degree to which existing regulations and guidance are affected and in need of revision, or new regulatory requirements established and new guidance developed, depends on the level of departure from existing fuel designs. The regulations at Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” provide principal design and performance requirements. The general design criteria (GDC) listed in Table 7.1 relate to fuel design and overall fuel performance under normal and accident conditions. Additional GDC may be affected if ATF performance becomes more challenging for the control or protection systems that ensure acceptable consequences under accident conditions. For each ATF design, the staff plans to map the hazards and failure mechanisms to the design and performance criteria of the GDC to determine the appropriate applicability and potential need for additional criteria.

Note that loading an ATF design in a specific plant will ultimately need to meet relevant plant specific criteria. This is especially important for those reactors in the United States that were licensed before the issuance of the GDC (about 40 percent of the operating plants).

Table 7.1 Potentially Affected GDC

GDC No.	Title
1	Quality Standards and Records
2	Design Bases for Protection against Natural Phenomena
10	Reactor Design
11	Reactor Inherent Protection
12	Suppression of Reactor Power Oscillations
13	Instrumentation and Control
20	Protection System Functions
25	Protection System Requirements for Reactivity Control Malfunctions
26	Reactivity Control System Redundancy and Capability
27	Combined Reactivity Control Systems Capability
28	Reactivity Limits
34	Residual Heat Removal
35	Emergency Core Cooling
61	Fuel Storage and Handling and Radioactivity Control
62	Prevention of Criticality in Fuel Storage and Handling

Even if a particular ATF design is unable to demonstrate verbatim compliance, the intent of these principal design and performance requirements should be satisfied or new requirements developed.

In addition to the GDC, the use of ATF may affect the regulations related to fuel design and performance listed in Table 7.2. For each ATF design, the staff plans to map the hazards and failure mechanisms to these requirements to determine whether any changes are necessary.

Table 7.2 Potentially Affected Regulations

Regulation (10 CFR)	Title
20	Standards for Protection against Radiation
50.34	Contents of Applications; Technical Information
50.46	Acceptance Criteria for Emergency Core Cooling Systems for Light -Water Nuclear Power Reactors
50.67	Accident Source Term
50.68	Criticality Accident Requirements
Part 50, Appendix B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
Part 50, Appendix K	ECCS Evaluation Models
Part 50, Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants
Part 100	Reactor Site Criteria

The regulatory guidance documents listed in Table 7.3 contain fuel -related information. For each ATF design, the staff plans to map the hazards and failure mechanisms to the guidance documents to determine what, if any, changes are necessary.

Table 7.3 Potentially Affected Guidance

Guidance Document	Title
NUREG-0630	Cladding Swelling and Rupture Models for LOCA Analysis
NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition
RG 1.157	Best -Estimate Calculations of Emergency Core Cooling System Performance
RG 1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors
RG 1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light -Water Nuclear Power Reactors
RG 1.203	Transient and Accident Analysis Methods

7.1 Additional Considerations

Aspects of ATF designs or implementation strategy such as the following could expand the scope, level of complexity, and schedule of the staff's review:

- an increase in uranium-235 enrichment, uranium-235 density, or fuel burnup beyond current limits for batch loading of ATF
- characterization of fission product release (e.g., chemical forms and release kinetics), core melt progression, core relocation, and mechanical and chemical interactions under severe accidents for non-UO_2 ceramic pellet fuel designs for the batch loading of ATF

The staff has recognized through heightened stakeholder interactions that requests for increased fuel burnup limits, beyond the current licensed limits, are very likely to be included along with near -term ATF applications. Therefore, the staff is proactively undertaking an initiative to begin assessing the current knowledge and experimental data base associated with high burnup fuels beginning with NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss -of -Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel" (ADAMS Accession No. ML013540584). Continued engagement with industry and the fuel vendors on this topic will inform the staff as to whether this plan needs to be amended to include the staff's complete strategy for addressing increases fuel burnup limits or if that activity can proceed in parallel with the plan.

Staff expects that industry decisions on targeted maximum burnups will direct plans with regard to an associated increase in enrichment to efficiently achieve the desired burnup. So along with the NRC staff's work associated with increased fuel burnups, the staff is also beginning an assessment of what enrichment increase the current knowledge and database could support. The NRC's project plan regarding higher fuel burnup and increased enrichment is discussed further in Appendix A of this document.

7.2 Lead Test Assemblies

LTA programs provide pool-side, post -irradiation examination data collection; irradiated material for subsequent hot -cell examination and research; and demonstration of in-reactor performance. This characterization of irradiated material properties and performance is essential for qualifying analytical codes and methods and developing the safety design bases for new design features or new fuel designs.

The NRC published a letter to the Nuclear Energy Institute on June 24, 2019, "Clarification of Regulatory Path for Lead Test Assemblies", (ADAMS Accession No. ML18323A169), that documents the agency's position with respect to the irradiation of ATF lead test assemblies (LTAs), and clarifies NRC's interpretation of when prior NRC approval is needed for LTA campaigns.

7.3 Initiating Activity

The staff's expenditures associated with developing regulatory strategies and the framework for design -independent ATF licensing began in fiscal year (FY) 2017 and will continue as long as DOE and industry actively pursue ATF development. The staff's expenditures to support design -specific regulatory hurdles will begin upon formal notification from a vendor of its intent to pursue licensing of a specific design.

7.4 Deliverables

Table 7.4 Anticipated In-Reactor Deliverables*

Title	Due Date (near term/longer term)
Map of hazards and failure mechanism to GDC, regulations, and guidance documents.	6–12 months from completion of the PIRT exercise
Develop or revise guidance to address any identified necessary changes.	24–48/36–60 months from completion of the PIRT exercise
Develop rulemaking to address any identified necessary changes.	24–48/36–60 months from identification of required change

* The technical lead is the NRR Division of Safety Systems, Nuclear Methods and Fuel Analysis Branch.

8 TASK 2: REGULATORY FRAMEWORK FUEL FACILITIES, TRANSPORTATION, AND STORAGE

8.1 Regulatory Infrastructure Analysis

The NRC gives the regulations for fuel cycle activities of fuel fabrication, radioactive material transportation, and spent fuel storage in 10 CFR Part 70, 10 CFR Part 71, and 10 CFR Part 72, respectively. The regulations identify general performance requirements and have been used for licensing a broad spectrum of fuel fabrication facilities and for the certification of a broad spectrum of transportation and storage packages. The NRC does not expect these regulations to need modification to accommodate the fabrication, transportation, or storage of ATF. Table 8.1.1 identifies the current guidance documents for the review of fuel facility licensing, transportation packages, and spent fuel storage designs.

Table 8.1 NRC Fuel Cycle Review Guidance

Review Guidance Document	Title
NUREG-1609	Standard Review Plan for Transportation Packages for Radioactive Material
NUREG-1617	Standard Review Plan for Transportation Packages for Spent Nuclear Fuel
NUREG-1520	Standard Review Plan for Fuel Cycle Facilities License Applications
NUREG-2215	Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities
Interim staff guidance	https://www.nrc.gov/reading-rm/doc-collections/isg/spent-fuel.html

These review guidance documents draw on industry experience in the fabrication, transportation, and storage of Zr-clad UO_2 fuel with up to 5-percent enrichment. The NRC may need to supplement some of the guidance to address safety -related issues that could arise from ATF designs that involve different fuel or clad materials, higher enrichment, or changes in the processes and systems used to produce or manage the ATF. Potential areas for which review guidance may be expanded include criticality safety for systems in which the enrichment is greater than 5 percent, fuel or cladding material properties that are used in the analysis of transportation or storage packages, and failure mechanisms that must be considered for irradiated fuel other than Zr-clad UO_2. Two specific examples for which guidance may be developed are material properties for FeCrAl alloys and SiC materials that are used as ATF cladding.

The NRC staff will continue to monitor industry plans for fabricating and transporting unirradiated ATF fuel designs and for managing irradiated ATF. When the staff believes that supplemental information or guidance would facilitate the preparation and review of applications involving the fabrication, transportation, and storage of ATF designs, it will discuss this with stakeholders and take actions where practical.

8.2 Facility, Transportation, and Storage Reviews

The regulatory reviews to support the development and deployment of ATF will occur in several fuel cycle areas over the near term to support irradiation of LTAs and over the longer term to support the batch deployment of ATF. The sections below discuss these various reviews.

8.2.1 Fuel Fabrication Facility Reviews

The ATF fabrication operations for UO_2-based fuel are expected to involve operations that are similar to currently licensed operations. Licensees will use the regulations at 10 CFR 70.72,

“Facility Change and Change Process,” to determine whether NRC approval is required before implementing a change for the production of ATF.

ATF fabrication operations that are substantially different from those used for the fabrication of Zr-clad UO_2 fuel (e.g., production of metal ATF, production or use of fuel material with enrichment greater than 5 percent) will likely require a license amendment. The NRC expects that licensees will submit such amendment requests at a later date, beyond the current planning horizon. Future updates of this plan will address such amendment requests as the industry’s plans become more certain.

8.2.2 Unirradiated Fuel Transportation Package Reviews

In the near term, the staff expects vendors that are developing ATF to request approval of packages for transporting LTAs from the fabrication facilities to reactors for test irradiation. Currently, two transportation package reviews for LTAs are planned as noted in Table 8.3. The staff will review these requests against the requirements of 10 CFR Part 71 and will use NUREG-1609 and pertinent interim staff guidance for the safety review.

As industry prepares for the batch loading of ATF, the staff expects to receive requests for the approval of transportation packages that allow large-scale (i.e., batch) shipment of unirradiated ATF assemblies. The staff expects that any such requests will be made after 2020 and that future updates of this project plan will address such activities more specifically as the industry’s plans become more certain.

The NRC staff will support PIRT efforts that focus on the identification and evaluation of material properties used in the safety analyses of transportation packages with ATF contents. These PIRT efforts are expected to help the staff develop additional regulatory guidance for ATF transportation, if required.

8.2.3 Irradiated Fuel Transportation Package and Storage Cask Reviews

The agency expects any shipments of irradiated ATF LTAs or rods from ATF LTAs to be made in NRC -approved transportation packages. Requests could be made under 10 CFR Part 71 (i.e., letters of special authorization) for a limited number of shipments of irradiated LTAs over a limited timeframe similar to that expected for unirradiated LTAs.

For the batch loading of ATF, the staff expects to receive requests for the approval of transportation packages under 10 CFR Part 71, which allows large-scale shipment of irradiated ATF assemblies. The staff expects that any such requests will be made after FY 2023, and future updates of this project plan will address such activities as the industry’s plans become more certain. The NRC will review these requests against the requirements of 10 CFR Part 71, and the staff will use NUREG-1617 and pertinent interim staff guidance for the safety review.

If NRC -licensed reactors use ATF assemblies, such plants will need storage systems for irradiated ATF that are licensed (or certified) under 10 CFR Part 72. The NRC expects the need for irradiated ATF storage systems to develop after 2023, and future updates of this project plan will address such systems as the industry's plans become more certain.

The NRC staff will support PIRT efforts that focus on the identification and evaluation of material properties and fuel degradation mechanisms that the review of transportation packages or storage systems for irradiated ATF should consider. These PIRT efforts should help the staff develop additional regulatory guidance for irradiated ATF, if required.

8.2.4 Potential Challenges

Certain aspects of ATF designs or fuel cycle implementation strategies could affect the scope, level of complexity, and schedule of the staff's review.

The major fuel cycle changes that are possible as a result of ATF development include (1) higher enriched uranium (e.g., greater than 5-percent enrichment), (2) different fuel material (e.g., Cr-doped UO_2, U_3Si_2, or metallic fuel material), and (3) different cladding (e.g., FeCrAl, SiC, or coated Zr cladding). The number and nature of changes in these areas affect the effort required to review proposed fuel cycle changes. Table 8.2 identifies potential regulatory actions for the fuel cycle facilities and operations that might be required for these potential fuel cycle changes.

Table 8.2 Potential ATF Fuel Cycle Action and Associated Regulatory Actions

Potential ATF Fuel Cycle Action	Potential Regulatory Actions at Affected Facilities/Operations			
	Enrichment Facility	Fuel Fabrication Facility	Transportation Operations	Irradiated Fuel Storage Facility
Higher enrichment	License amendment to produce higher enrichment material	License amendment to manufacture higher enriched fuel	Application for amendment to a transportation certificate or new transportation packages (fuel material) (e.g., uranium hexafluoride package)	Applications for amendments to a transportation certificate or new spent fuel storage systems are expected regardless of ATF enrichment (see box below)
Different fuel material		Facility changes that do not meet the criteria of 10 CFR 70.72(c) will require NRC approval	Application for amendment to, or new, transportation packages (unirradiated fuel, irradiated fuel)	Applications for amendments to, or new, spent fuel storage systems with ATF -specific license conditions
Different fuel cladding			Application for amendment to, or new, transportation packages (unirradiated fuel, irradiated fuel)	Applications for amendments to, or new, spent fuel storage systems with ATF -specific license conditions

The greater the differences between an ATF design and Zr-clad UO_2, the more likely supplemental review guidance will be required and the more likely the review will require greater staff effort. As an example, one potential ATF fuel material, U_3Si_2, is more susceptible to chemical reactions (e.g., water, air) than UO_2. This hazard needs to be considered in the design and operation of a facility that produces or stores this material, and the NRC staff will need to review such facility designs and safety controls as part of the licensing process.

8.2.5 Lead Test Assemblies

Limited near -term regulatory activities are expected for fuel cycle activities associated with the fabrication and transportation of LTAs that involve coated Zr-clad UO_2 fuel. Some regulatory actions may be necessary for the certification of transportation packages for LTAs that rely on structural performance of non-Zr cladding material.

8.2.6 Initiating Activity

The staff's expenditures associated with developing regulatory strategies and the framework for design in-dependent ATF fuel cycle licensing began in FY 2017 and will continue as long as DOE and industry are actively pursuing ATF development. The staff's expenditures to support design -specific fuel cycle regulatory hurdles will begin when responses to the ATF RIS identify future actions, when an applicant briefs the staff on its proposed submittal, or when the staff receives an application that presents details that it must review.

8.2.7 Anticipated Regulatory Actions

Near-term regulatory actions are all associated with the review of transportation packages for unirradiated LTAs, as identified in the Table 8.3. Other regulatory actions are expected and will be identified in future revisions of the plan after industry actions become clearer.

Table 8.3 Anticipated Fuel Cycle Regulatory Actions*

Anticipated Action	Assumed Submittal Due
Review of unirradiated LTA 1 transportation package	Summer 2018
Review of unirradiated LTA 2 transportation package	Fall 2018

* The technical leads are the (1) NMSS Division of Fuel Cycle Safety, Safeguards, and Environmental Reviews, Fuel Manufacturing Branch (fuel facilities), and (2) NMSS Division of Spent Fuel Management, Renewals and Materials Branch (transportation and storage).

9 TASK 3: PROBABILISTIC RISK-ASSESSMENT ACTIVITIES

The NRC uses probabilistic risk assessments (PRAs) to estimate risk to investigate what can go wrong, how likely it is, and what the consequences could be. The results of PRAs provide the NRC with insights into the strengths and weaknesses of the design and operation of a nuclear power plant. PRAs cover a wide range of NRC regulatory activities, including many risk -informed licensing and oversight activities (e.g., risk -informed technical specification initiatives, the significance determination process portion of the Reactor Oversight Process). These activities make use of both plant -specific licensee PRA models and plant -specific NRC PRA models. The NRC uses the former models predominantly for licensing and operational activities and the latter models predominantly for oversight activities. A key tenet of risk -informed decision -making is that these models reflect the as -designed, as -operated plant. For this reason, these models should be updated to reflect significant plant modifications. The introduction of significantly different fuel into the reactor core has the potential to affect these models, particularly once the reactor core composition significantly influences the plant's response to a postulated accident (e.g., time to fuel heat up and degradation, amount of total hydrogen generation).

Activities associated with the development of capabilities to support risk -informed regulatory activities following the implementation of ATFs could be significant, and information about the industry's intended approach is needed to create a meaningful plan. Early interactions within the PRA community on ATF activities, including early preapplication meetings, have been used to encourage industry to ensure that the approach being pursued is consistent with the related regulatory requirements and staff guidance. This plan recognizes that the staff's PRA -related preparatory work involves two separate, but closely related, aspects:

(1) The staff need to prepare for, and review, PRA -related information submitted as part of the licensing process for both the batch loading of ATF and incorporation of the safety enhancements of ATF into the licensing basis.

(2) The staff need to develop PRA -related capabilities that allow it to do the following effectively:

- Review risk -informed licensing applications and ensure that applicants are using acceptable PRA models once ATF is implemented.
- Perform risk -informed oversight evaluations (e.g., significance determination process) once ATF is implemented.

The nature of item 1 is highly dependent on the approach taken by each vendor or licensee, or both, in its licensing application. However, item 2 is somewhat independent of the licensing approach for the batch loading of ATF; therefore, this plan currently focuses more attention on item 2.

As illustrated by the above categorization, PRA is more broadly relevant to ATF than simply the incorporation of ATF safety enhancements into the licensing basis. Again, this stems from the fact that the NRC uses a risk -informed licensing and oversight approach that relies on plant -specific PRAs that represent the as -built and as -operated plant. Near -term ATF designs may have a limited impact on PRA modeling, whereas longer term ATF designs may have a more significant impact on PRA modeling. In general, the PRA modeling changes in question include the following:

- selection of core damage surrogates used in defining PRA end states (e.g., peak nodal clad temperature of 1,204 degrees Celsius, water level at two -thirds active fuel height)
- accident sequence modeling assumptions used to create event tree models that define the high -level successes and failures that can prevent core damage (e.g., late containment venting is required for avoiding core damage)
- system success criteria used in fault trees for defining the minimum hardware needed to fulfill specific mitigation functions (e.g., two relief valves are needed to prevent injection pump deadhead when feed and bleed cooling is used for a transient with no feedwater)
- sequence timing assumptions used in accident sequence modeling, success criteria determinations, and human reliability analysis to establish relevant time windows (e.g., feed and bleed cooling initiated within 20 minutes of -low steam generator water level).

The staff will need to ensure that licensees' PRAs continue to use acceptable models and assumptions as part of the implementation of ATF and update the NRC's models (as necessary) to reflect the ATF plant modifications. PRA models are not required under *Title 10 of the Code of Federal Regulations* (CFR) 50 and their use is not a prerequisite for approval of an ATF concept or its batch loading into a particular plant. That said, plants using PRA to support risk -informed operational programs (e.g., 10 CFR 50.69, Risk -Informed TS Initiatives) should continue to update their PRAs so that they realistically reflect the as -built, as operated plant. The NRC expects that modifications affecting a plant's risk profile (e.g., ATF, improved reactor coolant pump seals) will be incorporated into licensee's PRA models under their existing PRA maintenance programs.

Much of the needed underlying deterministic knowledge to address these points can leverage the work covered elsewhere in this plan, particularly the fuel performance, thermal hydraulics, and severe accident calculation capability development. It is envisioned that much of the analytical investigation needed to assess PRA related impacts and support PRA --related changes in the agency's SPAR models can use the MELCOR modeling and analysis discussed elsewhere in this plan. If needed, additional confirmatory analysis could also be pursued using MELCOR plant models developed for other NRC initiatives (such as those documented in NUREG-1953, "Confirmatory Thermal -Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Surry and Peach Bottom," issued September 2011, and NUREG-2187, "Confirmatory Thermal -Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Analysis Risk Models—Byron Unit 1," issued

January 2016). This leveraging of resources between severe accident analysis tools and PRAs is routine.

In the nearer term, PRA -related impacts can be assessed using the general knowledge being developed in these other ATF project plan areas in conjunction with one or more pilot efforts using the existing SPAR models. Such pilots would help gain risk insights, assess the potential changes in core damage frequency (CDF) and large early release frequency (LERF),[2] and highlight areas where existing guidance[3] or methods may require refinement to address the implementation of ATF.

As a final introductory point, engagement on PRA -related topics both within the staff and with external stakeholders is important at all stages. Effective interaction will foster a common understanding of the acceptability of PRA methods used to model plant modifications and the impact that will ultimately be realized when these modifications are integrated into PRAs and risk -informed processes. Effective interaction can also ensure that information required to develop PRA modeling assumptions related to plant modifications is properly coordinated with the deterministic review. In this case, PRA relevance has been identified early in the process, and time is available to address the PRA -related needs in a thoughtful and symbiotic manner.

For the purpose of identifying the PRA -related milestones, the following key assumptions are necessary (some restate assumptions made elsewhere in this plan):

- The timing of PRA -related efforts will be cross -coordinated with those of the previously identified partner areas (e.g., severe accident analysis) to allow the leveraging of deterministic work to make the PRA -related efforts efficient. A different approach might be needed if there is a strong desire to assess the industry's early perspective on the potential risk significance of ATF designs as they relate to future submittals aimed at leveraging ATF to reduce regulatory requirements.
- For all designs in question, the earliest TR/LAR review would start in 2020, with longer term ATF design licensing reviews occurring no earlier than 2023.
- This plan does not account for new regulatory initiatives that might be requested to maximize the operational or economic benefit of ATF, such as the following:
  - modifications to the categorization process in 10 CFR 50.69, "Risk -Informed Categorization and Treatment of Structures, Systems and Components for

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2 Differences in LERFs could occur because of (1) differing fuel heatup and degradation time windows, (2) the generation of differing amounts of in- vessel hydrogen, (3) changes to the fission product release rates, and (4) shifts in the balance of challenges to other vessel and connected piping system components stemming from higher in-core temperatures before the relocation of debris.

3 This guidance encompasses the guidance used in risk-informed licensing and oversight (e.g., the SRP; relevant RGs; Inspection Manual Chapter (IMC) 0609, "Significant Determination Process," dated April 29, 2015; the risk-assessment standardization process manual). In reality, most of this guidance would not require revisions because the concepts and processes would continue to apply. However, some aspects could require modifications, such as those involving the LERF multipliers used in IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," dated May 6, 2004, whereas some guidance may benefit from additional discussion of ATF impacts.

Nuclear Power Reactors," associated with the use of relative (as opposed to absolute) CDF/LERF criteria[4]

– reduction of requirements associated with security and emergency preparedness programs

Table 9.1 PRA Activities—Milestones

	Milestone	Input Needed	Lead Time/ Duration	Needed By
1	Participate in internal and external discussions and knowledge development related to ATF (e.g., internal working group meetings, public meetings)	N/A	Ongoing	N/A
2	Complete licensing reviews, including potential TRs or industry guidance, related to the risk -informed aspects of ATF licensing	More information regarding the specific licensing approach	TBD	TBD
3	Complete a SPAR pilot of a near -term ATF design for a boiling -water reactor (BWR) and pressurized -water reactor (PWR) subject plant to assess CDF/LERF impacts, gain risk insights, and identify potential improvements to guidance	Deterministic knowledge base being developed under other tasks (e.g., MELCOR analysis)	6 months	1 year before the first near -term ATF core load[1]
4	Complete a SPAR pilot of a longer -term ATF design for a BWR and PWR subject plant to assess CDF/LERF impacts, gain risk insights, and identify potential improvements to guidance	Deterministic knowledge base being developed under other tasks (e.g., MELCOR analysis)	6 months[2]	1 year before the first longer term ATF core load[1]
5	Update guidance (as necessary) to support licensing and oversight functions for plants making ATF -related modifications	Completion of the items above	1 year	Before the ATF core load[1]
6	Update agency PRA models to reflect ATF -related changes to the as -built, as -operated plant for relevant plants/models	Details of the plant modifications	1 year[3]	As needed to support the agency's risk evaluations

[1] Here, core load means the replacement of a large proportion (e.g., 50 percent or more) of the core with ATF assemblies, assuming that non -ATF fuel will be generally more limiting to PRA impacts if a mixed core exists.

[2] This task should be performed sequentially after the equivalent task for near -term ATF designs as long as both near -term and longer term designs are of regulatory interest.

[3] This would occur after approval of the associated licensing action.

[4] This initiative has been mentioned as a potential limitation in the degree of benefit that would be gained in risk-informed licensing space, and it contrasts to the use of absolute risk measures in other relevant risk-informed licensing activities such as risk-informed technical specification initiatives.

Table 9.2 PRA Activities—Deliverables*

Title	Lead Time
Safety Evaluation contributions for TRs and LARs related to ATF	TBD
Report that documents results and recommendations from a near -term ATF SPAR pilot study	1 year before the first near-term ATF core load
Report documenting results and recommendations from a longer -term ATF SPAR pilot study	1 year before the first longer term ATF core load
Updated guidance (e.g., risk -assessment standardization project guidance changes) to support licensing and oversight functions for plants making ATF -related modifications	Varies depending on the documents that require modifications
Updated agency PRA models to reflect ATF -related changes to the as -built, as -operated plant for relevant plants/models	As needed to support the agency's risk evaluations

* The technical lead is the NRR Division of Risk Analysis, Probabilistic Risk Assessment Licensing Branch.

10 TASK 4: DEVELOPING INDEPENDENT CONFIRMATORY CALCULATION CAPABILITIES

Independent confirmatory calculations are one of the tools that the staff can use in its safety review of TRs and LARs. Confirmatory calculations provide the staff insight on the phenomenology and potential consequences of transient and accident scenarios. In addition, sensitivity studies help to identify risk significant contributors to the safety analyses and assist in focusing the staff's review. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," identifies the standard format and content of safety analysis reports for nuclear power plants, and the SRP identifies the criteria that the staff should use to review licensee safety analyses. The NRC plans to continue to develop independent confirmatory analysis tools that support robust SEs and provide insights into safety significant factors for each ATF design. Vendor codes used for ATF modeling capabilities will likely be based on smaller data sets than those of the current Zr-UO_2 models. This will result in greater uncertainty in the results of the safety analyses and the margins to the specified acceptable fuel design limits. For these reasons, confirmatory calculation capabilities will be critical for generating confidence in the safety assessment of ATF against all applicable regulatory requirements (see Section 7 for more details). A confirmatory code can be used to independently quantify the impact of modeling uncertainties and support more efficient reviews with the potential for fewer requests for additional information. Finally, the experience and insights gained by developing an in-house code can be leveraged in reviews of externally developed models and methods, thus making reviews more efficient and effective.

The staff identified four technical disciplines needing calculation capability development to support TR/LAR safety reviews: (1) fuel performance, (2) thermal hydraulics, (3) neutronics, and (4) severe accidents. The NRC has developed a suite of codes to analyze these disciplines, and they have been used successfully to support regulatory decision -making. Further development of these codes is appropriate to ensure that the NRC has the capability to analyze ATF designs. Having tools that the staff can use to analyze ATF will be particularly important because applicants will use computational tools to demonstrate that they have met

fuel safety acceptance criteria and because, in some cases, the ATF properties and models within the computational tools will be based on limited experimental data.

The development of calculation capabilities will proceed with similar activities in each area, as follows:

- PIRT exercises help ensure that all new phenomena important to safety have been identified and considered in the planning phases. PIRT results will be used to inform code development efforts.
- Scoping studies will be performed to identify the architecture and model updates needed to model various ATF concepts.

- Where necessary, code architecture modifications will be made (e.g., to remove Zr/UO_2 hard wired properties and assumptions or to solve the governing equations for non-cylindrical geometry).
- Material properties will be added, and new models will be developed, where necessary.
- Integral assessment of the updated codes will be completed and documented. It is likely that results from integral assessments and uncertainty studies performed using updated codes will be used to revisit and maintain PIRT products.

Figure 10.1 depicts a generic schematic of tasks associated with developing calculation capabilities for near -term ATF designs, whether such capabilities are developed by the applicant, DOE, or the NRC. This figure defines the “lead time” and “duration” concepts for calculation capability. Lead time refers to the time required to complete the process for developing the calculation capability, and durations refer to the time required for the conduct of discrete tasks within the process.

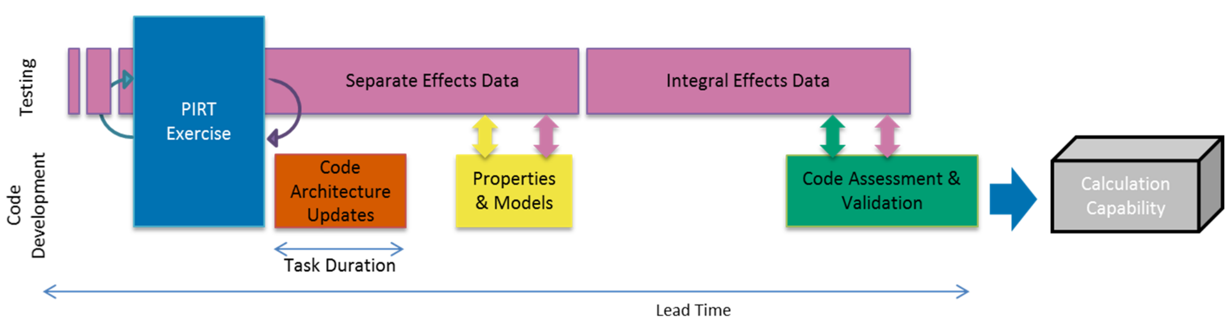

Figure 10.1 Development process for near -term ATF calculation capability

Figure 10.1 shows that code development requires testing and data to feed model development and validation. Developing codes to demonstrate that ATF can be used safely includes updating codes with ATF material properties and models and then validating the updated codes against relevant experimental data. The validation exercise ensures that a code appropriately models key phenomena and accurately predicts the parameters of safety importance. The datasets used to develop models often come from separate effects testing, whereas code assessment and validation often use data generated in integral test programs. The lead time to develop a calculation capability is intrinsically linked to the production and availability of data

from ongoing testing programs. The DOE and EPRI MOUs establish mechanisms that the NRC staff can use to communicate data needs discovered through model development and code assessment efforts.

The diagram is relevant for all ATF concepts; however, the NRC recognizes that some concepts have limited new phenomena and therefore the duration, and breadth, of each element will vary with each ATF concept.

The calculation capability development process for longer-term ATF designs will likely be more iterative than the process for near -term ATF designs and some of the tasks may proceed in parallel. This is because it is expected that some code architecture updates and new model needs will only become evident as more data for longer -term ATF designs becomes available and as codes are assessed. As with near -term ATF designs, the property and model development and the code assessment and validation tasks require data. Therefore, the duration of these tasks are intrinsically linked to the production and availability of data from ongoing testing programs.

For each discipline, the level of effort to complete these activities will vary based on the characteristics of the ATF design and the availability of information on the properties and phenomenological behaviors of the fuel, which will be addressed for each discipline in separate reference material. The estimated lead times to develop the codes necessary to analyze all currently proposed fuel/cladding types range from 3 to 6 years. The lead times include all code development activities and consider the time required to generate new data and new models for code development and integral assessment. The lead times vary by discipline and vary for near -term and longer -term ATF designs. Generally, longer lead times are estimated for longer term designs with the expectation that new phenomenological models will need to be developed and validated. The lead times are not independent between various ATF designs because it is anticipated that code architecture updates made for the first design can be leveraged for other ATF designs.

Although this plan addresses calculation capability development in four different disciplines, technical overlap between disciplines exists, including the introduction of new material properties. To reduce duplication of effort, the analysis tools will be coupled to allow codes to send and receive information between each other. For example, neutronics codes can be used to provide fuel performance codes with pellet radial power distribution information as a function of burnup, and fuel performance codes can provide neutronics codes with fuel temperature and deformation calculations. Thus, coupling the codes leverages information sharing to improve the overall analysis capabilities and ensures consistency across codes. The NRC will update its graphical user interface, Symbolic Nuclear Analysis Package, as needed to make it the interfacing tool between the NRC's suite of analysis tools. Where possible, the NRC will coordinate with DOE to reduce duplication of effort in calculation capability development.

Appendix B to this project plan describes the NRC's plans to develop analysis capabilities in the areas of fuel performance, neutronics, thermal hydraulics, and severe accidents and source terms.

11 PATH FORWARD

This project plan represents the high -level strategy to prepare the NRC for conducting efficient and effective reviews of ATF designs. The plan is intended to be a living document that may evolve as industry plans are refined and the state of knowledge for ATF concepts advances. The plan will be augmented with concept -specific licensing roadmaps at appropriate intervals to clearly identify the regulatory criteria which must be satisfied for approval.

The staff's priority, now that this plan has been finalized, is to: 1) engage directly with the nuclear fuel vendors pursuing near -term ATF concepts with the objective of understanding the nexus between the phenomena identified as important to safety and their testing plans, and 2) understand the areas of margin recovery or operational flexibility that licensees plan to seek such that staff can begin to proactively refine the regulatory framework where necessary.

APPENDIX A: FUEL BURNUP AND ENRICHMENT EXTENSION PREPARATION STRATEGY

Based on stakeholder interactions, the NRC staff is aware of industry's plans to request higher fuel burnup limits along with the deployment of near-term ATF concepts. Additionally, the staff expects that the extension of fuel burnup limits, and the economic drive to achieve those burnups, will result in requests to increase fuel enrichment to greater than the current standard of 5 weight percent uranium-235. Therefore, the staff is proactively assessing the current knowledge and experimental database associated with extending both burnup and enrichment for light water reactor (LWR) fuels. This plan focuses on the strategy to prepare the NRC for review of future licensing actions in which industry requests to go beyond current licensed limits with burnup up to ~75 gigawatt days per metric ton of uranium (GWd/MTU) rod-average and enrichment up to ~8 weight percent (wt%) uranium-235. Staff will continue to engage with industry and the fuel vendors on these topics and adjust this strategy as industry plans for higher burnup and increased enrichment evolve.

Overview of Preparatory Activities

As with other ATF activities related to advanced cladding and fuel materials, the staff has grouped its burnup and enrichment preparatory activities into four tasks. The highlights of each task are briefly described below; subsequent sections within this appendix describe these tasks in greater detail.

<u>Task 1: Regulatory Framework: In-Reactor Performance</u>

- Participate in coordinated Phenomenon Identification and Ranking Table (PIRT) exercises on in -reactor performance of fuels with increased enrichment under a wide array of conditions, performance -based metrics, and analytical criteria to ensure acceptable performance.
- Perform a scoping study to (1) evaluate the applicability of existing regulations and guidance for higher burnup and increased enrichment, (2) identify changes to existing regulations and guidance, or the need for new regulations and guidance, and (3) identify any key policy issues.
- Identify consensus standards that need to be updated for higher burnup and increased enrichment and participate in the update process where appropriate.
- Determine and clarify the regulatory criteria that need to be satisfied for higher burnup fuels and fuels with increased enrichment and the regulatory options available to applicants and vendors.
- If needed, resolve policy issues and initiate rulemaking and guidance development activities.

Task 2: Regulatory Framework: Fuel Cycle, Transportation, and Storage

- 10 CFR Part 70, “Domestic Licensing of Special Nuclear Materials” is a performance based; therefore, the staff does not anticipate identification of gaps or deficiencies in these regulations for the licensing of enrichment facilities to produce increased enrichment material or fuel fabrication facilities to fabricate increased enrichment fuel. The staff has previously licensed plants that produce uranium fuel enriched to the levels addressed in this plan.
- 10 CFR Part 71, “Packaging and Transportation of Radioactive Material”; and 10 CFR Part 72, “Licensing Requirements for the independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste, and Reactor- -Related Greater Than Class C Waste,” are largely performance based; therefore, the staff does not anticipate identification of gaps or deficiencies in these regulations.
- Gaps in the review guidance may develop as the fuel cycle industry develops plans for manufacturing, transporting, and storing higher burnup and increased enrichment fuel. The NRC staff will monitor the fuel cycle industry’s plans and identify and develop any necessary regulatory guidance in a timely manner.
- The NRC staff is engaging industry to understand the details and timing of its plans to produce uranium hexafluoride (UF_6) or other uranium precursor forms that are enriched above the current limit (5 weight percent uranium-235), and plans to fabricate increased enrichment LWR fuel.

Task 3: Probabilistic Risk Assessment- Activities

Like the impacts of ATF cladding and fuel matrix concepts, higher burnup and increased enrichment manifest in a probabilistic risk assessment (PRA) via impacts on the plant’s response to a postulated accident, in the form of changes to assumptions about sequence timing, success criteria, and severe accident phenomenology. The PRA activities described in the main body of this document (i.e., the activities originally crafted to address changes in plant response to beyond-design-basis accidents associated with ATF) may adequately encompass the PRA-related work needed to address the impacts of higher burnup and increased enrichment. The specific timeframes and nature of the industry activities and associated NRC deterministic technical basis development will dictate the degree of overlap between the two sets of activities. For instance, the pilot PRA model work described in Section 9 may be able to accommodate the potential burnup and enrichment changes combined with the other cladding and fuel response impacts associated with ATF. The degree of coverage provided by the pre-existing planning will also depend on the degree to which burnup and enrichment changes impact other agency uses of PRA information (such as in assessing environmental impacts associated with postulated accidents). At this time, the staff is assessing whether higher burnup and increased enrichment warrant any additional or different ATF-related PRA work, and the staff will adjust its planning accordingly.

Task 4: Developing Independent Confirmatory Calculation Capabilities

Independent confirmatory calculations are one of the tools that the staff can use in its safety review of topical reports and license amendment requests. Confirmatory calculations provide the staff insight on the phenomenology and potential consequences of transient and accident scenarios. In addition, sensitivity studies help to identify risk significant contributors to the safety analyses and assist in focusing the staff's review.

The staff's approach to modifying and validating existing NRC codes and performing confirmatory analysis for burnup and enrichment extension will be similar to the approach for ATF described in Section 6.4 in the ATF Project Plan. At this time, the NRC staff plans to modify existing NRC codes that are developed to analyze fuel performance, thermal hydraulics, neutronics, and severe accidents and source terms to support confirmatory analysis of fuels with higher burnup and increased enrichment. See Section 6.4 and Appendix B of the ATF Project Plan for further details.

A.1 Task 1: Regulatory Framework: In-Reactor Performance

Higher fuel burnup and increased enrichment present new and unique technical issues that current guidance, review plans, and regulatory criteria may not readily address. To prepare the agency to conduct meaningful and timely licensing reviews of higher fuel burnup and increased enrichment proposals, well developed- and vetted positions are needed on potential policy issues that may arise during the review and licensing process. These positions must be communicated to stakeholders clearly and early.

This task addresses the changes to the in-reactor regulatory framework that may be required to support the implementation of higher fuel burnup and increased enrichment considering the technical issues they present. Generally, the technical issues associated with higher fuel burnup and increased enrichment respectively fall into two categories, fuel integrity (cladding or fuel pellet) and nuclear criticality safety. ECCS performance embrittlement mechanisms and fuel fragmentation, relocation, and dispersal (FFRD) are examples of fuel integrity technical issues associated with higher burnup. Spent fuel pool criticality, and potential fast critical conditions during accident scenarios are examples of the technical issues associated with increased enrichment that fall under nuclear criticality safety. The regulatory framework changes that may be necessary to address each technical issue are likely to be different, and the staff anticipates that such changes will need to be made before either higher fuel burnup or increased enrichment can be predictably licensed for use outside of the exemption process.

The degree to which existing regulations and guidance need revision or new regulatory requirements and guidance need to be established, depends on the level of departure from existing burnup and enrichment limits. With regard to the regulations at Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” the NRC staff has concluded that the general design criteria (GDC) discussed therein will not be affected by higher burnup and increased enrichment. While higher burnup and increased enrichment may impact the way compliance with regulatory requirements is demonstrated, the actual principal design and performance requirements provided by the GDC remain applicable. Note that loading increased enrichment fuel designs in a specific plant will ultimately need to meet relevant plant-specific criteria. This is especially important for those reactors in the United States that were licensed before the issuance of the GDC (about 40 percent of the operating plants).

Beyond the GDC, higher burnup and the use of fuel with increased enrichment may affect the regulations and guidance related to fuel design and performance and nuclear criticality safety listed in Tables A.1 and A.2, below. The staff plans to map the technical issues and potential failure issues to these requirements and guidance to determine the scope of changes that are necessary.

Table A.1 Potentially Affected Regulations

Regulation (10 CFR)	Title	Affected by:	
		Burnup	Enrichment
50.34	Contents of Applications; Technical Information	✓	✓
50.46	Acceptance Criteria for Emergency Core Cooling Systems for Light Water- Nuclear Power Reactors	✓	✓
50.67	Accident Source Term	✓	✓
50.68	Criticality Accident Requirements		✓
50, Appendix I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents	✓	✓
50, Appendix K	ECCS Evaluation Models	✓	✓
51	Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions (specifically, Tables S-3 and S-4)	✓	✓
70.24	Criticality Accident Requirements		✓
100	Reactor Site Criteria	✓	✓

Table A.2 Potentially Affected Guidance

Guidance Document	Title	Affected by:	
		Burnup	Enrichment
NUREG-0630	Cladding Swelling and Rupture Models for LOCA Analysis	✓	
NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (Section 4.2, “Fuel System Design” in particular for burnup)	✓	✓
NUREG-1465	Accident Source Terms for Light-Water Nuclear Power Plants	✓	✓
NUREG-1555	Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan	✓	✓
NUREG-2121	Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident	✓	
NUREG/CR-7022 Vol. 1-2	FRAPCON-3.5	✓	✓
NUREG/CR-7023 Vol. 1-2	FRAPTRAN 1.5	✓	✓
NUREG/CR-7024	Material Property Correlations: Comparisons Between FRAPCON-3.5, FRAPTRAN 1.5, and MATPRO	✓	✓
NUREG/CR-7219	Cladding Behavior During Postulated Loss-of-Coolant Accidents	✓	
RG 1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	✓	✓
RG 1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	✓	✓
RG 1.203	Transient and Accident Analysis Methods	✓	✓
DG 1327	Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents	✓	✓

A.1.1 Additional Considerations

Aspects of higher burnup and increased enrichment fuel designs or the implementation strategy could expand the scope, level of complexity, and schedule of the staff’s review. Specifically, an increase in fuel burnup or uranium-235 enrichment could impact the scope of the staff’s environmental review and have implications for the license renewal generic environmental impact statement (GEIS) associated with a plant’s licensing basis.

Higher fuel burnup and increased enrichment may affect the NRC's generic environmental findings as documented in the GEIS. Licensees seeking to adopt either higher fuel burnup or increased enrichment beyond the current licensed limits will need to submit a license amendment request, and this submittal will need to provide sufficient information as to the potential environmental impacts of the request to facilitate the staff's review. The staff will need to review the environmental impacts of its actions, and this could be a source of additional complexity. To minimize this additional complexity, the staff may need to consider evaluating the environmental impacts generically, depending on the nature and volume of the requests. The necessity of this effort will become clearer as NRC staff continues engagement with industry and the fuel vendors.

Higher fuel burnup and increased enrichment may also effect changes in accident source term and operational source term via changes in decay heat load and isotopic inventory. Should these source terms be impacted, licensees will need to evaluate the impact of the change to the accident analyses and offsite doses, and may need to revise their accident analyses of record and environmental analyses. Additional challenges may exist if the revised source terms result in environmental impacts that are not captured in or bounded by the impacts discussed in the generic environmental impact statement (discussed above). This could complicate successful completion of a finding of no significant impact for an exemption request. The NRC staff will perform MELCOR calculations for representative plants to determine whether existing source term guidance (e.g., Regulatory Guide 1.183) remains applicable over the increased ranges of burnup and enrichment being proposed. The need to minimize the additional complexity that changes to source terms may pose, the manner in which such complexity may be addressed generically, and the necessity of these efforts will become clearer as NRC staff continues engagement with the industry and fuel vendors.

A.1.2 Lead Test Assemblies

Lead Test Assembly (LTA) programs provide poolside post- -irradiation examination data collection, irradiated material for subsequent hot cell- examination and research, and demonstration of in -reactor performance. This characterization of irradiated material properties and performance is essential for qualifying analytical codes and methods and developing the safety design bases for higher burnup fuels and fuels with increased enrichment.

The NRC has recently published a letter to the Nuclear Energy Institute (ADAMS Accession No. ML18323A169) that documents the agency's position concerning criteria for the insertion of LTAs under 10 CFR 50.59 without additional NRC review and approval. LTA programs for higher burnup and increased enrichment may require LARs, depending on the scope of the LTA campaign and the licensing basis of the reactor.

A.1.3 Licensing Strategy

The staff expects industry to take an incremental approach in moving to higher burnup and increased enrichment. Therefore, the NRC staff envisions near-term and longer-term strategies

for moving forward with the licensing of higher burnup fuels and fuels with increased enrichment. In the near-term, licensees will need to request exemptions to existing regulations on a licensee-specific basis for the use of either of these technologies and demonstrate compliance with safety requirements along with the exemption criteria. Should widespread adoption of these technologies become apparent, the NRC staff will utilize the longer-term strategy of rulemaking to update existing regulations to facilitate a more predictable licensing process.

A.1.4 Deliverables

The NRC staff plan to participate in coordinated PIRT exercises on the in -reactor performance of fuels with increased enrichment. The development of this PIRT will follow the Cr-coated cladding PIRT exercise example discussed in Section 3.4.3 of the ATF Project Plan (i.e., initial report on synthesis of public information, convene panel of experts, develop final report). At present, there is no plan to develop a separate PIRT for the in-reactor performance of fuels at higher burnup; the staff has concluded there is sufficient in-house expertise and data available within the literature to capture the in-reactor phenomenological response of conventional fuel designs at higher burnup (see, for example, NUREG/CR-6967 and NUREG-2121) and near-term ATF designs at higher burnup. The burnup performance of longer-term ATF designs is implicitly included in the respective PIRT for that design (see Section 3.4.3 of the ATF Project Plan). The dates listed below in Table A.3 are approximate, and they may need to be pushed back if there is insufficient data available to result in a work product of sufficient quality to validate a position or make a finding.

Table A.3 Anticipated In-Reactor Deliverables*

Title	Due Date	
	Burnup	Enrichment
Synthesize publicly available information and identify data gaps	End of 2019	9/2020
Complete PIRT panel exercises	N/A	6/2021
Final PIRT reports	N/A	9/2021
Map of technical issues and failure mechanisms to regulations, and guidance documents.	1/2020	12/2021 – 3/2022
Develop or revise guidance to address any identified necessary changes.	12/2021	12/2022 – 6/2023
Develop rulemaking to address any identified necessary changes.	24–36 months from identification of required change	24–36 months from identification of required change

* The technical lead is the office of Nuclear Reactor Regulation (NRR) Division of Safety Systems, Nuclear Performance and Code Review Branch

A.2 Task 2: Regulatory Framework: Fuel Cycle, Transportation, and Storage

Higher burnup and increased enrichment present different regulatory challenges throughout the fuel cycle. The NRC staff recognizes that these challenges have different timelines and that increased enrichment technical issues must be addressed in the near-term for successful deployment.

For the front end of the fuel cycle, which includes enrichment of the feed material, fuel assembly fabrication and transportation of feed material and fresh fuel assemblies, increased enrichment may present additional technical and regulatory issues; however, current guidance, review plans, and regulatory criteria are adequate to address these issues. To prepare the agency to conduct near-term licensing and certification reviews of increased enrichment levels, discussion of licensing and certification strategies and approaches between applicants and NRC staff will be undertaken to address any potential technical or policy issues that may arise. Any issues the NRC staff identifies will be communicated to stakeholders promptly.

For the back end of the fuel cycle, which includes transportation and storage of spent fuel at higher burnup and increased enrichment, the NRC staff will continue to monitor industry's initiatives and licensing actions for reactor operation, and assess whether revisions to current guidance, review plans, and regulatory criteria may be warranted. The NRC staff recognizes that licensing and certification actions related to the transportation and storage of such spent fuel will not occur in the near term. The NRC staff will engage with industry as plans on the back end of the fuel cycle are developed and will update this plan accordingly. Therefore, unless otherwise indicated, the rest of the discussion in this section will focus on near-term issues related to increased enrichment.

This task contemplates the changes to the regulatory framework that may be required to support the implementation of increased enrichment, considering the technical and regulatory issues it presents. When considering the safe transportation of material for the front end of the fuel cycle, the notable technical issue associated with increased enrichment pertains to nuclear criticality safety for UF_6 transportation and fresh fuel assemblies. Fuel assemblies (both fresh and irradiated) that rely on the fuel assembly structural performance to remain intact under accident conditions and the criticality evaluation of a single UF_6 package without using the exception in 10 CFR 71.55(g) are examples of the technical issues that fall under fuel integrity and nuclear criticality safety, respectively. Benchmarking criticality analyses for increased enrichment fuel and burnup credit analyses for spent fuel storage and transport are also examples of the technical issues that fall under nuclear criticality safety. The regulatory framework changes that may be necessary to address each technical issue are likely to be different; however, the staff does not anticipate that such changes will need to be made before higher fuel burnup or increased enrichment fuel can be licensed or certified for general use in reactors.

A.2.1 Regulatory Infrastructure Analysis

The regulatory requirements in 10 CFR Part 70, 10 CFR Part 71, and 10 CFR Part 72 govern the use of radioactive material for fuel enrichment and fabrication facilities, transportation, and spent fuel storage. For increased enrichment in UF_6 feed material and fresh fuel assemblies, changes to the regulations are not necessary to accommodate industry plans; however, licensing and certification challenges may exist. The criticality regulations in 10 CFR 71.55(g) grant an exception from the consideration of moderation intrusion for the transportation of UF_6 enriched to 5 weight percent or less. Transportation of UF_6 enriched to greater than 5 weight percent will require the design and certification of new packages, the modification of currently approved packages, or an exemption from the regulations that require evaluation of a single package with optimum moderation for enrichments greater than 5 weight percent.

Table A.4 identifies the current guidance documents for the review of fuel facility licensing, transportation package certification, and spent fuel storage licensing and certification and identifies whether the guidance document is affected by industry plans to use increased enrichment or higher burnup fuel.

Table A.4 NRC Fuel Cycle Review Guidance

Review Guidance Document	Title	Affected By	
		Burnup	Enrichment
NUREG-1609[1]	Standard Review Plan for Transportation Packages for Radioactive Material	✓	✓
NUREG-1617[1]	Standard Review Plan for Transportation Packages for Spent Nuclear Fuel	✓	✓
NUREG-1520	Standard Review Plan for Fuel Cycle Facilities License Applications		✓
NUREG-2214	Managing Aging Processes In Storage (MAPS) Report	✓	✓
NUREG-2215	Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities	✓	✓
NUREG-2224	Dry Storage and Transportation of High Burnup Spent Nuclear Fuel	✓	✓
Spent Fuel Storage and Transportation Interim staff guidance[2,3]	https://www.nrc.gov/reading-rm/doc-collections/isg/spent-fuel.html	✓	✓

[1] Note that NUREG-1609 and NUREG-1617 are being combined into a single standard review plan, NUREG-2216, "Standard Review Plan for Transportation Package Approval," which is scheduled to be completed in the summer of 2020.

[2] After completion of NUREG-2215 and NUREG-2216, all existing Interim Staff Guidance documents issued by the Division of Spent Fuel Management will be retired.

[3] In particular, SFST-ISG-8, Revision 3, "Burnup Credit in the Criticality Safety Analysis of PWR Spent Fuel in Transport and Storage Casks," is affected by both higher burnup and increased enrichment.

These review guidance documents draw on industry experience in the fabrication, transportation, and storage of Zr clad- UO_2 fuel with up to 5 weight percent enrichment and burnup up to approximately 62 GWd/MTU rod average. The NRC staff may need to supplement existing guidance to address safety-related issues associated with increased enrichment and higher burnup. If NRC staff believes that supplemental information or guidance would facilitate the preparation and review of applications involving the enrichment, fabrication, transportation, and storage of either higher burnup or increased enrichment fuel, it will discuss this with stakeholders and take action where practical.

A.2.2 Facility, Transportation, and Storage Reviews

The regulatory reviews to support the development and deployment of increased enrichment fuel will occur in several fuel cycle areas over the near term to support production (enrichment and fuel fabrication) and transportation of UF_6 feed material and fresh fuel assemblies. The sections below discuss these various reviews.

A.2.2.1 Uranium Enrichment and Fuel Fabrication Facility Reviews

The uranium enrichment facilities that produce enriched uranium as well as fabrication operations that would produce conventional fuel (e.g., Zr-clad UO_2) with increased enrichment will conduct operations that are similar to currently licensed operations. These licensees will have to submit amendments to produce or use uranium with increased enrichment. Fuel fabrication operations that use new processes for producing a different type of fuel material (e.g., uranium alloy or U_3Si_2) are expected to submit amendments to address both increased enrichment as well as the new processes.

The staff is currently engaged with licensees of fuel cycle facilities to understand the status of their plans and the anticipated timing of their license amendment submittals.

A.2.2.2 Uranium Feed Material and Unirradiated Fuel Transportation Package Reviews

As industry prepares for the batch loading of increased enrichment ATF, the staff expects to receive requests for the approval of transportation packages that allow large-scale (i.e., batch) shipment of uranium feed material (currently UF_6) and unirradiated fuel assemblies. The staff will review these requests against the requirements of 10 CFR Part 71 and will use NUREG-1609 and pertinent interim staff guidance for the safety reviews. The NRC staff will support PIRT efforts that focus on criticality safety (criticality code validation) and materials properties and performance of increased enrichment fuel. These PIRT efforts are expected to help the staff develop additional regulatory guidance for transportation of fuel with increased enrichment, if required.

The staff is currently engaged with fuel cycle facility certificate holders to understand the status of their plans and the anticipated timing of their certificate amendment submittals.

A.2.2.3 Irradiated Fuel Transportation Package and Spent Fuel Storage Reviews

The back end of the fuel cycle—spent fuel storage and transport—presents some challenges that are similar to the front end. For example, benchmarking criticality safety is still an issue for the back end for enrichments between 5 and 8 weight percent, but additional challenges may exist depending on the licensing or certification strategy. Other areas where challenges may exist include performance of the cladding material during vacuum drying, aging while in dry cask storage, fatigue data for transportation, and benchmarking the isotopic depletion analyses for use in the shielding analyses for higher burnup fuels and for use in burnup credit criticality analyses.

The staff is currently engaged with fuel cycle facility certificate holders to understand the status of their plans and the anticipated timing of their certificate amendment submittals.

A.2.2.4 Potential Challenges

The NRC staff has identified technical challenges for transportation of unirradiated fuel with increased enrichment and spent fuel with higher burnup and increased enrichment.

A.2.2.4.1 Challenges for Transportation of Unirradiated Fuel

In addition to challenges for approval of transport of UF_6 at increased enrichment (greater than 5 weight percent), it should be noted that American National Standards Institute (ANSI) N14.1, "Nuclear Materials — Uranium Hexafluoride – Packagings For Transport," only applies to enrichments up to 5 weight percent uranium-235 for the 30B and 30C cylinders. DOT regulations in *Title 49 of the Code of Federal Regulations* (CFR) 173.420 state that UF_6 packaging (whether fissile, fissile excepted, or non-fissile) must be designed, fabricated, inspected, tested and marked in accordance with American National Standard N14.1 that was in effect at the time the packaging was manufactured. DOT regulations in *Title 49 of the Code of Federal Regulations* (CFR) 173.417, which provide requirements for shipment of UF_6 heels without a protective overpack also limit the enrichment of 30B and 30C cylinders to 5 weight percent. In addition to an NRC approval for shipment in a packaging using a 30B or 30C cylinder, a special permit from DOT will be needed.

Benchmarking criticality analyses for fissile material enriched to greater than 5 weight percent uranium-235 presents a challenge due to the limited number of critical experiments in that range. Applicants for package approval could potentially overcome this challenge by:

- performing new critical experiments to validate criticality calculations for 5-8 wt% enriched uranium,
- relying on sensitivity/uncertainty analysis methods to develop new critical experiments,
- relying on sensitivity/uncertainty analysis methods to determine that existing experiments are applicable to 5-8 wt% enriched uranium-235,

- increasing the one-sided k-effective tolerance factor to account for uncertainties in criticality code performance due to the number of applicable critical experiments for benchmarking, or
- using some combination of the above options.

A.2.2.4.2 Challenges for Transportation and Storage of Spent Fuel

In addition to the benchmarking challenge listed above, other challenges exist for the storage and transportation of spent fuel. Evaluation of material performance during vacuum drying, aging while in storage, and cladding material properties are needed to evaluate structural performance during normal storage, transport, and accident conditions.

Aging effects during long-term, dry cask storage include evaluation of impacts of potential operable age-related phenomena on cladding performance. Those mechanisms described in NUREG-2214 that may be affected by higher burnup and increased enrichment include creep, hydrogen absorption, oxidation, delayed hydride cracking, and irradiation hardening. In addition, the impacts of both potential higher end-of-life rod internal pressures on the credibility of age-related phenomena, and the increased pellet swelling on the mechanical performance of the cladding should be evaluated.

The NRC staff has also historically expected that experimental confirmation be obtained and assessed which confirms that the spent fuel performs as expected. The experimental confirmatory basis that low-burnup fuel (≤45 GWd/MTU) remains in its analyzed configuration during the period of extended operation was provided in NUREG/CR-6745, “Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination” (Bare and Torgerson, 2001), and NUREG/CR-6831, “Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage” (Einziger et al., 2003). The research results in NUREG/CR-6745 and NUREG/CR-6831 support a determination that degradation of low-burnup fuel cladding and assembly hardware should not result in changes to the approved design bases during the first period of extended operation, provided that the cask/canister internal environment is maintained. The NRC staff expects that similar experimental data be obtained to confirm that unanalyzed age-related phenomena are not at play during the dry storage and subsequent transport of spent fuel with high burnup and increased enrichment.

Regarding transportation of spent fuel with higher burnup and increased enrichment, fatigue performance data will be needed to evaluate vibration normally incident to transport as required in 10 CFR 71.71(c)(5).

A transportation package or storage cask that is evaluated containing spent fuel will have the same benchmarking concerns listed above for unirradiated material. If a package or cask is evaluated for burnup credit, instead of fresh fuel, the isotopic depletion analyses will need to be validated for the increased enrichment and burnup levels. In addition to validating the criticality analysis, the accuracy of depletion calculations to calculate the source term for the shielding analyses should be evaluated for burnup greater than 62 GWd/MTU rod average.

However, these challenges would not preclude an effective and efficient staff review.

A.2.2.5 Anticipated Regulatory Actions

Near term- regulatory actions consist of reviews of fuel cycle facilities license amendments. At present, only one fuel cycle facility has shared plans to submit a license amendment. Other expected regulatory actions will be identified in future revisions of the plan after industry plans become clearer.

A.2.3 Deliverables

The NRC staff plan to participate in coordinated PIRT exercises that focus on criticality safety (criticality code validation) and materials properties and performance of increased enrichment fuel. These PIRT efforts are expected to help the staff develop additional regulatory guidance for transportation of fuel with increased enrichment, if required. The development of these PIRTs will follow the discussion in Section 3.4.3 of the ATF Project Plan. The dates listed below in Table A.5 are approximate, and they may need to be pushed back if there is insufficient data available to result in a work product of sufficient quality to validate a position or make a finding.

Table A.5 Anticipated Fuel Cycle, Transportation, and Storage Deliverables*

Title	Due Date	
	Burnup	Enrichment
Initial report on transportation of feed material and fresh fuel with increased enrichment	N/A	9/2020
Initial report on material properties/performance and storage of fuel with increased enrichment at higher burnup, including criticality concerns	3/2021	3/2021
Complete PIRT panel exercises	6/2021	6/2021
Final PIRT reports	9/2021	9/2021

* The technical lead is the office of Nuclear Material Safety and Safeguards.

A.3 Task 3: Probabilistic Risk Assessment Activities

The NRC staff uses PRAs to estimate risk: to investigate what can go wrong, how likely it is, and what the consequences could be. The results of PRAs provide the NRC staff with insights into the strengths and weaknesses of the design and operation of a nuclear power plant. PRAs cover a wide range of NRC regulatory activities, including many risk informed licensing and oversight activities (e.g., risk- informed technical specification initiatives, the significance determination process portion of the Reactor Oversight Process). These activities make use of both plant- specific licensee PRA models and plant- -specific NRC PRA models. The NRC staff uses the former models predominantly for licensing and operational activities and the latter models predominantly for oversight activities. A key tenet of risk informed decision- making is that these models reflect the as- designed, as- -operated plant. For this reason, these models should be updated to reflect significant plant modifications. The introduction into the reactor core of fuels intended for higher burnup and fuels with increased enrichment may affect these models, particularly once the reactor core composition significantly influences the plant's response to a postulated accident (e.g., higher initial decay heat from increased unrainum-235 enrichment).

Developing capabilities to support risk informed- regulatory activities following the implementation of higher fuel burnup and increased enrichment could require significant NRC resource. Information about the industry's intended approach is needed to create a meaningful plan. Early NRC staff interactions with the industry and vendors regarding higher burnup and increased enrichment activities, such as fuel technology update meetings and early preapplication meetings, will be used to encourage an approach that is consistent with regulatory requirements and staff guidance. Just as with the ATF Project Plan, this project plan recognizes that the staff's PRA -related preparatory work involves two separate, but closely related, aspects:

(1) The staff needs to prepare for, and review, PRA related- information submitted as part of the licensing process for the batch loading of fuels with increased enrichment and higher burnup as well as the incorporation of these technologies into the licensing basis.

(2) The staff needs to develop PRA related- capabilities to do the following effectively:

- Review risk informed- licensing applications and ensure that applicants are using acceptable PRA models once higher fuel burnup and increased enrichment are implemented.
- Perform risk informed- oversight evaluations (e.g., significance determination process) once higher fuel burnup and increased enrichment are implemented.

Item 1 is highly dependent on the approach taken by each vendor or licensee, or both, in its licensing application, while item 2 is somewhat independent of the licensing approach. Therefore, this project plan currently focuses more attention on item 2.

Incremental increases in fuel burnup and enrichment (such as increases on the orders of tenths of a percent enrichment or ones of gigawatt days per metric ton) would have only a limited (or no) impact on PRA modeling. However, the more appreciable increases in fuel burnup and enrichment that are anticipated, especially in combination with the other cladding and fuel changes associated with adoption of ATF, would have a more significant impact on PRA modeling.

PRA activities for higher burnup and increased enrichment will be analogous to the activities for ATF described in Section 9 of this document. In particular, NRC staff must ensure that licensees' PRAs continue to use acceptable models and assumptions as part of the implementation of higher burnup fuels and fuels with increased enrichment and update the NRC's models (as necessary) to reflect any plant modifications made to accommodate these new technologies. Also analogous to the activities for ATF, it is envisioned that much of the analytical investigation needed to assess PRA related impacts and support PRA- -related changes in the agency's SPAR models due to higher burnup or increased enrichment can use the independent confirmatory calculational capabilities currently being developed by the NRC staff. These capabilities are discussed in Section A.4 of this project plan. See Section 9 of this document for further information on the analogous PRA activities NRC staff will take in response to higher burnup and increased enrichment.

Engagement on PRA -related topics both among the NRC staff and with external stakeholders is important at all stages. Effective interaction will foster a common understanding of the acceptability of PRA methods used to model plant modifications and the impact that will ultimately be realized when these modifications are integrated into PRAs and risk -informed processes. Effective interaction can also ensure that information required to develop PRA modeling assumptions related to plant modifications is properly coordinated with the deterministic review. In this case, the relevance of PRAs has been identified early in the process, and time is available to address the PRA -related needs in a thoughtful and informed manner.

For the purpose of identifying PRA related- milestones, the following key assumptions are necessary:

- The timing of PRA related efforts will be cross- coordinated with those of the previously identified partner areas (e.g., severe accident analysis) to allow the leveraging of deterministic work to make the PRA- -related efforts efficient.
- Near-term TR/LAR reviews will start in 2020, with long-term licensing reviews occurring no earlier than 2023.
- This plan does not account for rulemaking initiatives that might be requested to facilitate rapid adoption of increased enrichment (e.g., modifications to 10 CFR 50.68, "Criticality Accident Requirements").

The PRA-related milestones for higher burnup and increased enrichment activities are listed below in Table A.6. It should be noted that it may be feasible to merge the work outlined in Table A.6 with the existing ATF PRA-related milestones found in Table 9.1, depending on the nature and timing of the higher burnup and increased enrichment activities relative to that of the ATF activities.

Table A.6 PRA Activities for Higher Burnup and Increased Enrichment—Milestones

	Milestone	**Input Needed**	**Lead Time/ Duration**	**Needed By**
1	Participate in internal and external discussions and knowledge development related to higher burnup and increased enrichment (e.g., internal working group meetings, public meetings)	N/A	Ongoing	N/A
2	Complete licensing reviews, including potential TRs or industry guidance, related to the risk -informed aspects of licensing higher burnup fuels and increased enrichment	More information regarding the specific licensing approach	TBD	TBD
3	Complete a SPAR pilot of a BWR and PWR subject plant for higher burnup and increased enrichment to assess CDF/LERF impacts, gain risk insights, and identify potential improvements to guidance	Deterministic knowledge base being developed under other tasks (e.g., independent confirmatory code analysis)	6 months	1 year before the first long term- core load[4] of higher burnup fuels and fuels with increased enrichment
4	Update guidance (as necessary) to support licensing and oversight functions for plants making modifications (if any) to accommodate higher burnup and increased enrichment	Completion of the items above	1 year	Before the core load
5	Update agency PRA models to reflect changes to the as built, as- -operated plant (if any) for relevant plants/models	Details of the plant modifications	1 year[5]	As needed to support the agency's risk evaluations

[4] Here, core load means the replacement of a large proportion (e.g., 50 percent or more) of the core.

[5] This would occur after approval of the associated licensing action.

Table A.7 PRA Activities for Higher Burnup and Increased Enrichment—Deliverables*

Title	Lead Time
Safety Evaluation contributions for TRs and LARs related to using fuels with higher burnup and increased enrichment	TBD
Report that documents results and recommendations from a SPAR pilot study	1 year before the first long term- core load of higher burnup fuels and fuels with increased enrichment
Updated guidance (e.g., risk assessment- standardization project guidance changes) to support licensing and oversight functions for plants making modifications (if any) to accommodate higher burnup and increased enrichment	Varies depending on the documents that require modifications
Updated agency PRA models to reflect changes to the as -built, as -operated plant (if any) for relevant plants/models	As needed to support the agency's risk evaluations

* The technical lead is the NRR Division of Risk Analysis, Probabilistic Risk Assessment Oversight Branch.

A.4 Task 4: Developing Independent Confirmatory Calculation Capabilities

Independent confirmatory calculations are one of the tools that the staff can use in its safety review of topical reports (TRs) and license amendment requests (LARs). Confirmatory calculations provide the staff insight on the phenomenology and potential consequences of transient and accident scenarios. In addition, sensitivity studies help to identify risk significant contributors to the safety analyses and assist in focusing the staff's review.

RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," identifies the standard format and content of safety analysis reports for nuclear power plants, and NUREG-0800, "Standard Review Plan for the Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) identifies the criteria that the staff should use to review licensee safety analyses. The NRC staff plans to continue to develop independent confirmatory analysis tools that support robust safety evaluations and provide insights into safety significant factors for higher burnup fuels and fuels with increased enrichment. Vendor codes used to support analysis of fuel above existing burnup and enrichment limits will likely be based on smaller data sets than the data sets available for Zr-UO_2 fuel below existing limits. This will result in greater uncertainty in the results of the safety analyses and the margins to the specified acceptable fuel design limits. For these reasons, confirmatory calculation capabilities will be critical for generating confidence in the safety assessment of burnup and enrichment extension against all applicable regulatory requirements (see Section A.1 and Section A.2 for more details). A confirmatory code can be used to independently quantify the impact of modeling uncertainties and support more efficient reviews with the potential for fewer requests for additional information. Finally, the experience and insights gained by developing an in-house code can be leveraged in reviews of externally developed models and methods, thus making reviews more efficient and effective.

The staff identified four technical disciplines needing calculation capability development to support TR/LAR safety reviews for higher burnup and increased enrichment: (1) fuel performance, (2) thermal hydraulics, (3) neutronics, and (4) severe accidents. The NRC staff has developed a suite of codes to analyze these disciplines, and they have been used successfully to support regulatory decision -making. Further development of these codes is appropriate to ensure that the NRC staff has the capability to analyze Zr-UO_2 fuel above existing regulatory burnup and enrichment limits. Having tools that the staff can use to analyze fuel with higher burnup or increased enrichment will be particularly important because applicants will use computational tools to demonstrate that they have met fuel safety acceptance criteria and because, in some cases, the properties and models for fuel at higher burnup or increased enrichment within the computational tools will be based on limited experimental data.

Code development activities for higher burnup and increased enrichment will be integrated and sequenced, as appropriate, with activities for ATF described in Section 10 of the ATF Project Plan. In particular, the NRC staff will participate in PIRT exercises for increased enrichment, perform scoping studies to identify code architecture and model updates needed, modify the codes based on outcomes of the increased enrichment PIRT and scoping studies, and perform

assessments against available experimental data. Section 10 of the ATF Project Plan describes the approach NRC staff will take to update its codes to support confirmatory analysis for higher burnup and increased enrichment limits.

APPENDIX B: NRC PLANS TO DEVELOP ANALYSIS CAPABILITY

Task 4.a: Fuel Performance

The U.S. Nuclear Regulatory Commission's (NRC's) longstanding fuel performance codes, FRAPCON and FRAPTRAN, have been merged into a single, modern code called "FAST" (Fuel Analysis under Steady-state and Transients), which will need to be updated to analyze the performance of accident tolerant fuel (ATF) and support licensing reviews. The NRC staff uses fuel performance codes during licensing reviews to demonstrate that specified acceptable fuel design limits are maintained and to provide initial conditions for design -basis accident (DBA) analysis. Additionally, the staff uses fuel performance codes to support the safety limits for loading and storing spent nuclear fuel in dry casks. Updates to fuel performance codes needed to model some ATF designs, including iron -chromium -aluminum (FeCrAl) cladding and coated zirconium (Zr)-based alloy claddings, will require minimal changes to FAST, with work focusing on new material properties, code assessment, and benchmarking. More extensive updates are needed to model silicon carbide (SiC) tubing, non -uranium -dioxide (UO_2) fuel, and fuel designs in noncylindrical fuel forms.

FAST development activities include the following tasks:

- **Scoping Study**—For FAST, this will be a lower -source, straightforward activity because the code development needs are largely understood.
- **Code Architecture Updates**—This includes modifying the FAST code to be more modular. The staff will remove any computational assumptions embedded in the code for the Zr/UO_2 system and ensure that the heat transfer, solid mechanics, and diffusion solutions are generic while calling a separate set of libraries (MatLib) containing the relevant material properties. The staff will modify FAST to allow for modeling noncylindrical fuel forms, which entails re-solving the physics modeled by the code (e.g., heat conduction, diffusion, solid mechanics) with different geometrical conditions and possibly modeling in multidimension. Finally, infrastructure development will include changes necessary to allow for FAST to interact with other NRC tools via the Symbolic Nuclear Analysis Package (SNAP) interface. Additionally, FAST is currently under development to support non -light-water -reactor licensing activities, including the development of multidimensional solvers, which are expected to reduce the level of effort in the code architecture updates for longer term ATF concepts.
- **Property and Model Development**—Material properties (nonirradiated and irradiated) for ATF claddings and fuels will be added to MatLib (the material properties library in FAST) and models developed to address new phenomenon and failure modes presented by ATF. A series of thermal, mechanical, and irradiation -induced properties needs to be updated for each new fuel and cladding material. For new fuels, additional considerations include fission gas release and fuel creep. For claddings, additional correlations are needed to model hydrogen pickup, steady -state and transient corrosion, thermal and irradiation creep, high -temperature deformation, and new failure models. As experience is gained through the in -reactor use of ATF fuel and cladding, the staff

expects that significant data will be available to the NRC to develop models that capture the evolution of the properties as a function of burnup (e.g., thermal conductivity degradation). The staff expects that additional properties will be needed to model new materials that are yet unknown, such as the diffusion of oxygen or volatiles and fuel/cladding/coolant interaction, which may require more extensive modifications to both MatLib and FAST. As the phenomena become known, models related to long -term spent fuel handling, storage, and transportation will be updated.

- **Code Assessment and Validation**—Integral performance data from the Advanced Test Reactor, Halden, Transient Reactor Test Facility, and lead test assembly programs, as well as other available sources, will be used to confirm that the material properties and models added to FAST fully account for the integral behavior of ATF. This will be the most time -consuming task for several reasons. The licensing requirements for each ATF design will determine the focus of code assessment. For example, the current assessment of FAST for steady -state calculations looks at fission gas release, fuel centerline temperature, cladding strain, oxidation, and rod internal pressure, all of which are part of the specified acceptable fuel design limits outlined in Section 4.2, "Fuel System Design," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP). The code assessment, also referred to as the integral assessment, analyzes the integral effects of all the models and correlations working together to analyze the thermal -mechanical behavior of the fuel rod under typical light -water -reactor conditions. A proper assessment requires numerous cases that cover the breadth of boundary conditions and operating regimes that the fuel design will experience under normal operation, anticipated operational occurrences, and DBA conditions. For example, the FAST integral assessment currently consists of more than 200 nonproprietary cases for the UO_2/Zr system and numerous proprietary cases and data sets that the code and its predecessors have been assessed against over the last several decades. The integral assessment is the key to identify phenomena that may not be properly modeled. If discrepancies are shown between the code and the data, then the code's models and correlations will be reexamined and updated as necessary to achieve reasonable agreement. This will require an iteration between architecture updates (if new physics are determined to occur), material properties updates, and rerunning the integral assessment. In addition, the more detailed the integral assessment, the more knowledge is gained on understanding the uncertainties of the code. The amount of data that are available to create the assessment database greatly affects the uncertainty of the results of analyses from the fuel performance codes. As more data are included in the assessment database, the confidence in the results of the analyses increases. As the size of the assessment database increases, topical report (TR) reviews will become more efficient because the results of the fuel performance codes will have less uncertainty.

The tables below list the milestones for this task, with their related trigger or needed input, lead time, and schedule driver.

Table 4.a.1 Near -Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Low level of resources needed; short -duration task		
Code Architecture Updates	-	1 year	-
Material Property and Model Development	Separate effects data	2 years	-
Code Assessment and Validation	Integral effects data	1 year	Fuel TR submittal
Lead Time for Fuel Performance Calculation Capability for Near -Term ATF	2–4 years		

Table 4.a.2 Longer Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Low level of resources needed; short -duration task		
Code Architecture Updates (Remove Zr and UO_2 hard-wired properties)	-	1 year[1]	-
Code Architecture Updates (Remove assumptions related to fuel geometry and Zr-UO_2 interaction)	Completion of previous milestone	2 years[2]	-
Material Property and Model Development	Separate effects data available	2–4 years[2]	-
Code Assessment and Validation	Integral effects data	2–3 years	Fuel TR submittal
Lead Time for Fuel Performance Calculation Capability for Longer Term ATF	3–5 years		

[1] Task will not be required if the near -term activities are completed.
[2] Tasks can be worked in parallel.

Technical Lead: Office of Nuclear Regulatory Research (RES)/Division of Systems Analysis (DSA)/Fuel and Source Term Code Development Branch (FSCB)

Task 4.b: Thermal -Hydraulics

To support confirmatory analyses for licensing reviews, the NRC's TRAC/RELAP Advanced Computational Engine (TRACE) system safety thermal -hydraulic code will need to be updated to analyze ATF fuel performance. TRACE will support the performance of design -basis transient and DBA analyses and sensitivity studies. TRACE uncertainty quantification tools help assess the impact that uncertainties in material properties have on fuel performance in DBAs. The existing TRACE fuel rod model assumes that the fuel is oxide fuel and the cladding is a zirconium alloy. ATF designs may have different fuel or cladding or both. Therefore, updates to the TRACE fuel rod models will be needed to model the ATF designs. Small changes to TRACE should be able to accommodate cylindrical fuel with metallic cladding, including FeCrAl cladding and coated zirconium -based alloy claddings. More extensive updates would be needed to model SiC tubing, any non-UO_2 fuel, and metallic fuel in noncylindrical fuel forms. TRACE may also need to be coupled to external fuel rod models such as FAST or the U.S. Department of Energy's (DOE's) BISON to model some aspects of ATF. The expected limiting factor on completion is obtaining adequate data for materials and testing from DOE or fuel vendors.

This task involves the following activities:

- **Scoping Study**—A scoping study will be conducted to determine the changes needed to allow TRACE to perform plant accident and transient calculations with ATF. A set of sample plant calculations will be selected to demonstrate the changes in plant response as a result of ATF.
- **Code Architecture Updates**—TRACE will need updates to the fuel and cladding mechanical and thermal material properties for the new or different materials proposed for use in ATF designs. Additionally, the cladding oxidation and rupture models in TRACE are based on empirical data for zirconium -alloy cladding and will need to be modified to model clad oxidation and rupture for the other cladding materials proposed for use in ATF designs. In the case of noncylindrical solid metallic fuel rods, a method will be needed to analyze the noncylindrical geometry and the impact it has on conduction and convective heat transfer and fluid flow. TRACE may also need to be coupled to external fuel rod models such as the FAST or the BISON fuel analysis models through the TRACE Exterior Communications Interface to model some aspects of ATF.
- **Property and Model Development**—The physical models that need to be updated are fuel and cladding mechanical and thermal properties, cladding oxidation kinetics models, clad rupture models, fuel boiling and convective heat transfer models (including minimum stable film boiling temperature models for each cladding), and critical heat flux (CHF) models. The staff expects that the industry or DOE will provide most of this information.

- **Code Assessment and Validation**—The updated code will be validated against steady-state and transient data collected by DOE and the industry. Demonstration calculations of reactor accidents will be performed to examine the impact of the new fuel on the evolution of the accident and safety criteria.

The tables below list the milestones for this task, with their related trigger or needed input, lead time, and schedule driver.

Table 4.b.1 Near -Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Low level of resources needed; short -duration task, assuming models maintain same qualitative form as current models		
Code Architecture Updates	Assumes models maintain same qualitative form as current models; need conceptual design information (geometry, materials)	2 years	3 years before safety analysis TR submittal
Material Property and Model Development	Data available from literature, fuel vendor, and DOE test programs, etc.	2 years	2 years before safety analysis TR submittal
Code Assessment and Validation and Sample Plant Calculations	Reactor physics capability to generate cross sections/point kinetics parameters for sample plant calculations; data available from fuel vendor and DOE test programs	1.5 years	Assessment data needed 1.5 years before safety analysis TR submittal
Lead Time for Thermal -Hydraulic Calculation Capability for Near -Term ATF	3 years		

Table 4.b.2 Longer Term ATF Concepts[1]

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Low level of resources needed; short-duration task, assuming models maintain same qualitative form as current models		
Code Architecture Updates (Remove Zr and UO_2 hard-wired properties)	Conceptual design information for ATF designs (geometry, materials)	1 year[2]	-
Code Architecture Updates (Remove assumptions related to fuel geometry and Zr-UO_2 interaction)	Completion of previous milestone	1 years[3]	-
Material Property and Model Development	Separate effects data	1 years[3]	-
Code Assessment and Validation	Integral effects data	1.5 years[3]	Assessment data needed 1.5 years before safety analysis TR submittal
Lead Time for Thermal -Hydraulic Calculation Capability for Longer Term ATF	2 years		

[1] Table assumes near -term fuel work has been completed.
[2] Task will not be required if the near -term activities are completed and new models have the same qualitative form as the current models.
[3] Tasks can be worked on in parallel, assuming that the new models have the same qualitative form as the current models.

Technical Lead: RES/DSA/CRAB

Task 4.c: Neutronics

Neutronics calculations are an integral part of the confirmatory review process, as in, for example, NUREG-0800, Chapter 4, "Reactor," and Chapter 15, "Transient and Accident Analyses," because they provide decay heat rates, core power, and reactivity values used by thermal hydraulic and fuel performance codes. Neutronics analysis is also needed for the quantification of nuclide inventory for severe --accident/consequence analyses required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," and for evaluations supporting 10 CFR 50.68, "Criticality Accident Requirements"; 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"; and 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High -Level Radioactive Waste, and Reactor -Related Greater than Class C Waste." Neutronics analysis is also performed to support decisions for spent fuel pool loading and for confirming fluence calculations necessary to quantify vessel embrittlement, shielding analyses to support "as low as reasonably achievable" objectives, calculations of spent fuel pool decay heat rates and dose for human reliability analyses, core power/reactivity for transient calculations, assembly decay heat rates for cask loading, and other activities.

The NRC's main neutronics codes are SCALE, which provides a computational capability to evaluate nuclear systems, and GenPMAXS/PARCS to evaluate time -dependent core performance. The NRC staff uses SCALE to support licensing reviews by performing criticality safety evaluations of enrichment and fuel fabrication facilities, developing lattice physics parameters for reactor operations, and performing safety evaluations for transport and storage. The staff also uses SCALE for analyses of spent fuel pools and severe accidents and for input into probabilistic risk assessments. PARCS is used as a core simulator that supports thermal -hydraulic reviews under design -basis scenarios. GenPMAXS reads the lattice physics parameters from SCALE (nodal averaged cross sections and kinetic parameters) and converts the nuclear data into the format that is required by PARCS.

This task involves the following activities:

- **Scoping Study**—The current NRC neutronics packages, SCALE and GenPMAXS/PARCS, will be reviewed to understand the needed modifications, if any, to characterize ATF for the whole fuel cycle. These codes will be reviewed against the unique features of the ATF fuel designs (e.g., coated zirconium, doped UO_2, FeCrAl, uranium silicide, SiC) and for enrichments of higher than 5 weight percent uranium-235 (up to 20 weight percent). The SCALE code suite covers the functional areas of nuclear data and methods, Monte Carlo methods, isotope decay, depletion and activation methods, reactor physics methods, and sensitivity and uncertainty methods. The GenPMAXS/PARCS code covers reactor operations and transient performance.

  The scoping study will consider the needs for each of these functional areas. As such, it will involve a review of the fuel cycle and the associated impact of ATF designs.

- **Code Architecture Updates**—The required infrastructure development activities for SCALE and GenPMAXS/PARCS will include any code modifications and enhancements identified in the scoping study for implementation.

  Expected development activities include the following:

  - coupling of SCALE and PARCS with FAST in order to receive detailed fuel data (such as temperatures and geometry) while providing intra-pin radial power profiles and axial rod power profiles to FAST (to eliminate the need for FAST to develop new correlations)
  - SCALE geometry enhancements to support noncylindrical fuel for both three -dimensional Monte Carlo analysis and two -dimensional lattice physics calculations
  - SCALE input and modeling investigations and enhancements to model coated cladding, where coat thickness is less than 100 microns
  - updates to the energy group structures (both the fine multigroup structure that is used in SCALE and the collapsed broad group structure that is used by PARCS for core calculations) and to the nuclear data library, nuclear methods development (at the lattice and nodal level) to enable pin power reconstruction for noncylindrical fuel, and input interfaces

- **Model Development**—The required development activities for SCALE and GenPMAX/PARCS will include any code modifications and enhancements identified in the scoping study for implementation.

  Expected development activities include the following:

  - evaluation of the depletion and activation effects for ATF fuel and cladding compositions
  - evaluation of the impact of thermal -hydraulic and fuel performance effects that may be more or less important for ATF, such as thermal expansion, heat capacity, thermal conductivity, and swelling and gap closure
  - evaluation of a different cross section parameterization methodology and the development of different fuel temperature averaging techniques in order to better characterize Doppler feedback for the different carbide, ceramic, and metallic fuels under consideration
  - evaluation of the methodology for thermal -hydraulic calculations with steady -state PARCS calculations (PARCS includes PATHS, a simplified drift flux formulation, for thermal -hydraulic feedback), including the accommodation for more extreme axial discontinuities and heterogeneities, the evaluation of the coupling of the fuel and wall temperature to the determination of bulk fluid temperature, and the evaluation of the applicability of the current constitutive relationships (void fraction quality models, subcooled quality, wall friction factor, and two-phase flow friction factors)

- sensitivity and uncertainty assessment, including assessment of modeling techniques, for ATF candidates
- evaluation of tritium release through advanced cladding materials such as SiC

(Parts of these activities can begin before the receipt of necessary data.)

- **Code Assessment and Validation**—Code verification and validation are important elements of the SCALE and GenPMAXS/PARCS software quality assurance program.

  All new updates, modification, and enhancements must be assessed against test data. These would include test data from post irradiation examinations (e.g., destructive assay of fuel or clad to validate depletion) but would be finalized by the gap analysis.

  The code assessment and validation task also includes testing the combined code sequence for a particular application. Within the reactor operations space, codes such as SCALE and GenPMAX, and PARCS and TRACE, will be tested together, for example.

The tables below list the milestones for this task, with their related trigger or needed input, lead time, and schedule driver.

Table 4.c.1 Near -Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Short -duration task		
Code Architecture Updates	Completion of previous milestone	1 year	-
Model Development	Data as required from scoping study	1 year	-
Code Assessment and Validation	Data as required from scoping study	1 year	Safety analysis submittal
Lead Time for Neutronics Calculation Capability for Near-Term ATF	1–2 years		

Table 4.c.2 Longer Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Short-duration task		
Code Architecture Updates	Completion of previous milestone	2 years[1]	-
Model Development	Data as required from scoping study	2 years[1]	-
Code Assessment and Validation	Data as required from scoping study	2 years[1]	Safety analysis submittal
Lead Time for Neutronics Calculation Capability for Longer Term ATF	2–3 years		

[1] Tasks can be worked on in parallel.

Technical Lead: RES/DSA/FSCB

Task 4.d: Source Term

Several fuel vendors, in coordination with DOE, have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance. Certain ATF designs could lead to a departure from the current regulatory source terms used by the staff for fission product release during an accident, anticipated operational occurrences, or normal operation. The current source terms are based on insights derived from current generation light -water reactors using typical zirconium -alloy fuel. Prompted by these investigations into new fuel designs, the staff examined the technical bases of the various regulatory source terms to assess the potential impacts to the current assumptions of the regulatory process.

Regulatory source terms are deeply embedded in the NRC's regulatory policy and practices as the current licensing process has evolved over the past 50 years. The licensing process is based on the concept of defense in depth, in which power plant design, operation, siting, and emergency planning comprise independent layers of nuclear safety. This approach encourages nuclear plant designers to incorporate several lines of defense in order to maintain the effectiveness of physical barriers between radiation sources and materials from workers, members of the public, and the environment in operational states and, for some barriers, in accident conditions. The approach centers on the concept of DBAs, which aims to determine the effectiveness of each line of defense. The DBAs establish and confirm the design basis of the nuclear facility, including its safety -related structures, systems, and components and items important to safety, ensuring that the plant design meets the safety and numerical radiological criteria set forth in regulations. From this foundation, specific safety requirements have evolved through a number of criteria, procedures, and evaluations, as reflected in the regulations, regulatory guides, standard review plans, technical specifications, and license conditions, and as well as TID, WASH, and NUREG documents.

The various regulatory source terms, used in conjunction with the DBAs, establish and confirm the design basis of the nuclear facility, including items important to safety, ensuring that the plant design meets the safety and numerical radiological criteria set forth in the CFR (e.g., 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance"; 10 CFR 50.67, "Accident Source Term"; 10 CFR 50.34(a)(1)(iv); General Design Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", and subsequent staff guidance. When existing regulatory requirements, guidance, and procedures were developed, ATF designs that are being considered were not contemplated. Potentially impacted regulatory requirements, guidance, and procedures include the following:

- regulations (10 CFR Part 50; 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"; and 10 CFR Part 100)
- regulatory guides
- technical specifications
- emergency preparedness procedures
- evaluation methods for assessing the environmental impact of the accident

The SRP contains specific examples of the various regulatory source terms and provides information on the staff's regulatory guides. The various regulatory source terms discussed in guidance:

- Accident source term is based on DBAs to establish and confirm the design basis of the nuclear facility and items important to safety while ensuring that the plant design meets the safety and numerical radiological criteria set forth in the CFR (e.g., 10 CFR 100.11, 10 CFR 50.67, 10 CFR 50.34(a)(1)(iv), GDC 19, and subsequent staff guidance). SRP Chapter 15 addresses this topic.
- Equipment qualification source term is used to assess dose and dose rates to equipment. SRP Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment"; SRP Section 12.2, "Radiation Sources"; Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"; and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Appendix I, address this topic.
- Post accident shielding source term is used to assess vital area access, including work in the area. SRP Section 12.2; Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980; RG 1.89; and RG 1.183 address this area.
- Design -basis source term is based on 0.25–1-percent fuel defects to determine the adequacy of shielding and ventilation design features. SRP Section 12.2 provides further guidance.
- Anticipated operational occurrences source term is based on the technical specifications or the design -basis source term, whichever is more limiting, to determine the effects of events like primary -to -secondary leakage and reactor steam source term. SRP Section 11.1, "Coolant Source Terms," gives reactor coolant (primary and secondary) and reactor steam design details.
- Normal operational source term is based on operational reactor experience, as described in American National Standards Institute/American National Standard N18.1, "Selection and Training of Nuclear Power Plant Personnel." SRP Section 11.1 and Section 11.2, "Liquid Waste Management System," give further guidance for reactor coolant (primary and secondary) and reactor steam design details, and SRP Section 11.3, "Gaseous Waste Management System," gives system design features used to process and treat liquid and gaseous effluents before being released or recycled.

The NRC staff has concluded that an ongoing process is the appropriate method for incorporating new information on ATF -specific accident source terms. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in the applicable guidance, based on and justified by design -specific features. Regulatory Position 2 of Regulatory Guide 1.183 provides attributes of an acceptable alternative source term.

This task involves the following activities:

- **Scoping Study**—A review of the current capabilities of the MELCOR code is needed to better understand the necessary code modifications in the code packages for the simulation of accident progression (i.e., core heatup and degradation, combustible gas generation, and fission product release and transport to the containment).
- **Code Architecture Updates**—The required infrastructure development activities for MELCOR involve code modifications identified in the scoping study for the implementation of new or improved models. MELCOR contains various models for the modeling of the core components (e.g., fuel, cladding, and channel boxes), hydrogen and carbon monoxide generation and combustion, and fission product release from the core components. Some models and correlations (for fission product release and core degradation) in the code need to be modified for application to new fuel designs (e.g., through access to sensitivity coefficients and control functions or through generalized models). Examples include fuel rod collapse and eutectic interactions (e.g., impact of Zr-chromium intermetallic reactions) and the oxidation kinetics of cladding based on experimental data (at high temperature and pressure). The material properties for new designs also need to be added to the code database.
- **Property and Model Development**—The staff expects that clad coatings would affect oxidation behavior/combustible gas generation rate, and that fuel composition would affect fission product release rates and potentially chemical speciation. Other potential effects may exist, such as chemical reactions between fuel and clad that accelerate or retard core degradation or between fission products and clad or fuel that could either enhance or diminish the release of fission products. The code should be able to account for fission product speciation under various conditions, and information from experiments on fission product release (for non-UO_2 fuel only) is needed for the development of the source term and code assessment.

The MELCOR code is well suited for developing a regulatory source term with flexible models to support the evaluation of differences between current standard fuel and ATF concepts. The MELCOR models are general in nature and can be adjusted to reflect differing properties of advanced fuels. Depending on the results, the ATF models would likely be similar to those for standard fuel but with different parameters. Containment combustible gas control would need to be based on the expected release from the fuel/cladding combination given the clad type. For Lightbridge (or other future designs using metal fuel), the contribution of the fuel itself to gas generation may need to be considered.

- **Code Assessment and Validation**—Code verification and validation are important elements of the MELCOR software quality assurance program. All new code models need to be assessed against available test data. Experiments characterizing clad oxidation and combustible gas generation rates, fission product release magnitudes, and chemical forms and rates are required to ensure that the release models for ATF are representative and that such effects are accounted for, if significant.

- **Source Term Development**—The process for developing radiological source terms involves gathering the experimental data, developing and implementing applicable models in the MELCOR code, simulating a series of accidents representing most of the core damage frequency for both boiling -water reactors (BWRs) and pressurized -water reactors (PWRs) to obtain fission -product -group -specific release behavior (i.e., “gap” and early in-vessel fission products release initiation time and duration, core release fraction and chemical forms), and finally collapsing the data into a simplified representative set of release fractions and timings for rapid use in simplified codes for siting evaluation. For example, the source term analysis for high burnup and mixed --oxide fuel[1] was time consuming and involved about 30 detailed code calculations for BWRs and PWRs representing various accident scenarios (e.g., station blackouts, small- and large -break loss -of -coolant accidents).

The tables below list the milestones for this task, with their related trigger or needed input, lead time, and schedule driver.

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1 SAND2011-0128, “Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel,” issued January 2011.

Table 4.d.1 Near -Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Short -duration task		
Code Architecture Updates	Completion of previous milestone	1 year[1]	-
Model Development	Separate effects data	1 year[1]	-
Code Assessment and Validation	Integral effects data	1 year[1]	-
Lead Time for MELCOR Calculation Capability for Near -Term ATF	1 year		
Accident Progression Calculations with MELCOR	MELCOR code	1 year[2]	Safety analysis submittal
Lead Time for Source Term Development for Near -Term ATF	2 years[3]		

[1] Tasks can be worked on in parallel (involves different phenomenology).
[2] Task can be worked on after completion of MELCOR calculation capability.
[3] Includes the time for MELCOR capability and calculations.

Table 4.d.2 Longer Term ATF Concepts

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Short-duration task		
Code Architecture Updates	Completion of previous milestone	1 year[1]	-
Model Development	Separate effects data	2 years[1]	-
Code Assessment and Validation	Integral effects data	2 years[1]	-
Lead Time for MELCOR Calculation Capability for Longer Term ATF	2–3 years		
Accident Progression Calculations with MELCOR	MELCOR code	2–3 years[2]	Safety analysis submittal
Lead Time for Source Term Development for Longer Term ATF	4–6 years[3]		

[1] Tasks can be worked on in parallel (benefits from work done for near -term designs).
[2] Task can be worked on after completion of MELCOR calculation capability.
[3] Includes the time for MELCOR capability and calculations.

Technical Lead: RES/DSA/FSCB

APPENDIX C: CHANGE HISTORY

ITEM	LOCATION	REVISION	DESCRIPTION
1	Page 5, Section 2, Figure 2.1	1.1	ATF Steering Committee figure updated to reflect Office merger related changes.
2	Page 7, Section 3, Table 3.1	1.1	ATF Milestone Schedule table updated.
3	Page 13, Section 3.4.3	1.1	Section updated to reflect completed PIRT actions.
4	Page 25, Section 7.2	1.1	LTA section updated to identify agency position letter.
5	Page 25, Section 7.4, Table 7.4	1.1	Basic edits made to the table.
6	Appendix A	1.1	New Appendix A added: "Fuel Burnup and Enrichment Extension Preparation Strategy." Minor edits also made throughout document to capture the Appendix referencing.
7	Appendix B	1.1	Previous Appendix A moved to Appendix B. Minor editorial changes throughout.
8	Appendix C	1.1	New Appendix C added to capture document change history.