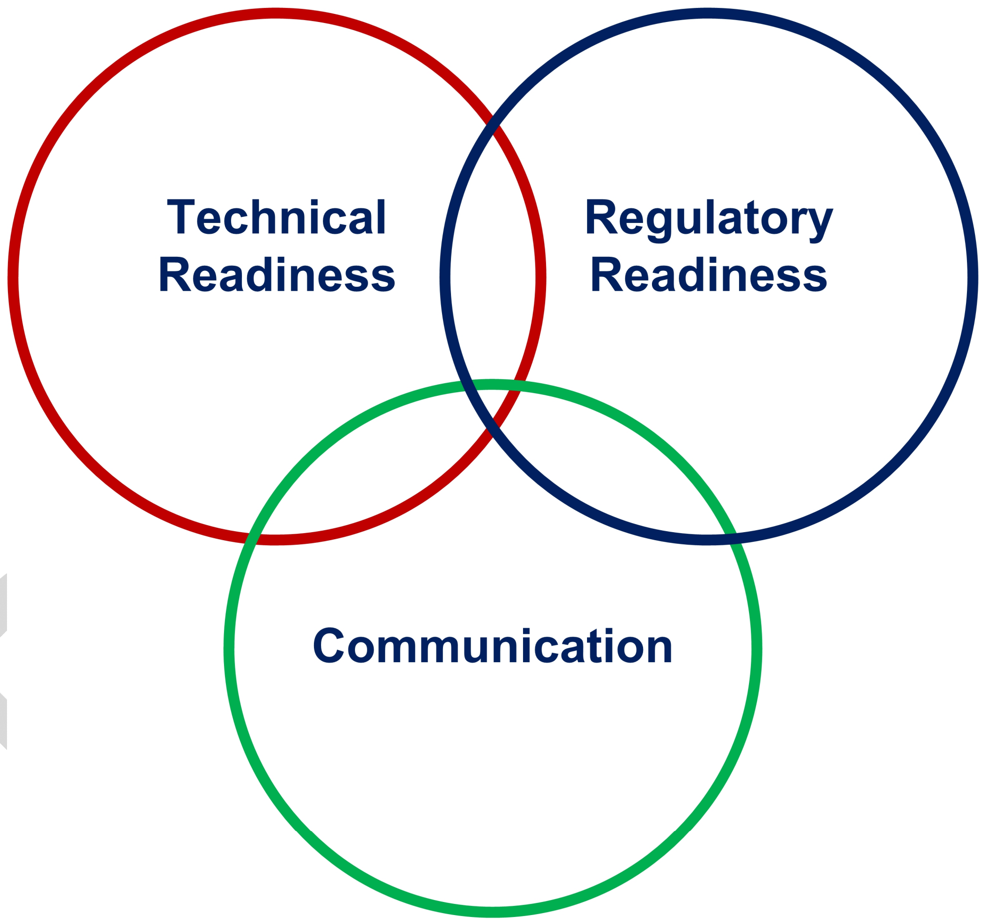
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy - Staff Report: Near-Term Implementation Action Plans



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Volume 2 – Detailed Information

1.0 INTRODUCTION

This volume (Volume 2) of the non-light water reactor (non-LWR) near-term implementation action plan (IAP) report provides the detailed, actionable steps and resource requirements necessary to support the NRC's strategic objectives of enhancing technical readiness, optimizing regulatory readiness, and optimizing communications. An overview of the NRC's vision and strategy for non-LWRs and an executive summary of the near-term readiness activities are presented in Volume 1 of the report.

This staff report covers the readiness actions to be taken in the next five years (the near-term activities), and will be supplemented with the mid-term and long-term plans in early 2017. The report provides the detailed IAPs for each strategyand extensive background information as needed to assist staff assigned to execute these plans. The staff has also developed accompanying estimates of job hours and contract support costs at the office and FY levels for internal budgeting and planning use. The staff will engage stakeholders to solicit feedback on this document and plans to finalize the near-term IAPs in early 2017.

2.0 SUMMARY – NEAR-TERM STRATEGIES AND CONTRIBUTING ACTIVITIES

This list summarizes the strategies and associated near-term contributing activities found in this report.

Strategy 1: Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory reviews

- Contributing Activity No. 1.1: Identify Non-LWR Task and Technical Skill Requirements (Work Design Activities)
- Contributing Activity No. 1.2: Determine and Establish the Necessary Workforce Skills and Capacities (Workforce Design & Establishment)

Strategy 2: Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews

Functional Area: Reactor Kinetics and Criticality

- Contributing Activity No. 2.1: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in the following operating High Temperature Gas-cooled Reactor (HTGR) modes (start-up; quasi-steady state cycle-specific operation; and transient analysis from a limiting point in cycle or equilibrium cycle).
- Contributing Activity No. 2.2: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.3: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in the following operating Sodium-Cooled Fast Reactor (SFR) modes (start-up; quasi-steady state cycle-specific operation; and transient analysis from a limiting point in cycle or equilibrium cycle).
- Contributing Activity No. 2.4: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.5: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in an operating Molten Salt Reactor (MSR), for steady state and transient analysis.
- Contributing Activity No. 2.6: Identify experimental data needs and begin code assessment.

Functional Area: Fuel Performance

- Contributing Activity No. 2.7: Develop knowledge of fuel design, fuel functional requirements, and fuel characteristics critical to safety and accident performance.
- Contributing Activity No. 2.8: Develop or adopt/update existing fuel analysis code applicable to HTGRs.
- Contributing Activity No. 2.9: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.10: Develop knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance.

- Contributing Activity No. 2.11: Develop or adopt/update existing fuel analysis code applicable to SFRs.
- Contributing Activity No. 2.12: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.13: Develop knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance.
- Contributing Activity No. 2.14: Develop fuel analysis code applicable to MSRs.

<u>Functional Area: Thermal-Fluid Phenomena</u>

- Contributing Activity No. 2.15: Develop thermal-fluid analysis code applicable to gas-cooled reactors.
- Contributing Activity No. 2.16: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.17: Develop thermal-fluid analysis code applicable to sodium-cooled fast reactors.
- Contributing Activity No. 2.18: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.19: Develop thermal-fluid analysis code applicable to molten salt reactors.
- Contributing Activity No. 2.20: Identify experimental data needs and begin code assessment.

<u>Functional Area: Severe Accident Phenomena</u>

- Contributing Activity No. 2.21: Develop severe accident analysis code applicable to gas-cooled reactors.
- Contributing Activity No. 2.22: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.23: Develop severe accident analysis code applicable to liquid metal-cooled fast reactors.
- Contributing Activity No. 2.24: Identify experimental data needs and begin code assessment.
- Contributing Activity No. 2.25: Develop severe accident analysis code applicable to molten salt reactors
- Contributing Activity No. 2.26: Identify experimental data needs and begin code assessment.

<u>Functional Area: Offsite Consequence Analysis</u>

- Contributing Activity No. 2.27: Perform an initial scoping study identifying and prioritizing potentially relevant modeling needs. Note: mid-term activity, included for information only.
- Contributing Activity No. 2.28: Based on the initial scoping study and design information available to date, implement needed modeling enhancements to be able to analyze offsite consequences for non-LWRs. Note: mid-term activity, included for information only.

<u>Functional Area: Materials and Component Integrity</u>

- Contributing Activity No. 2.29: Assess the performance needs and issues for structural materials to be used in non-LWRs, such as HTGR, SFR, MSR. The assessment will include the state-of-the-knowledge, ongoing domestic and international research, applicable international Operational Experience (OpE), codes and standards activities, gaps in knowledge, data, and assessment tools.
- Contributing Activity No. 2.30: Conduct research activities to develop technical bases to resolve major materials related issues. Collaborate with domestic (Department of Energy (DOE), Electric Power Research Institute (EPRI), vendors) and international regulatory partners [based on the recommendations from the assessment report from contributing Activity No. 2.29].
- Contributing Activity No. 2.31: Support the development of a draft regulatory framework for materials-related issues (relevant Standard Review Plan (SRP) chapters, guidance, etc.) for non-light water reactors.

Strategy 3: Develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes

- Contributing Activity No. 3.1: Establish and document the criteria necessary to reach a safety, security, or environmental finding for non-LWR applicant submissions. The criteria and associated regulatory guidance are available to all internal and external stakeholders.
- Contributing Activity No. 3.2: Determine and document appropriate non-LWR licensing bases and accident sets for highly prioritized non-LWR technologies.
- Contributing Activity No. 3.3: Identify, document and resolve (or develop plan to resolve) current regulatory framework gaps for non-LWRs.
- Contributing Activity No. 3.4: Develop and document a regulatory review “roadmap” that reflects the design development lifecycle and appropriate points of interaction with the NRC, and references appropriate guidance to staff reviewers and applicants.
- Contributing Activity No. 3.5: Prepare and document updated guidance for prototype testing, research and test reactors.
- Contributing Activity No. 3.6: Engage reactor designers and other stakeholders regarding technology- and design-specific licensing project plans and develop regulatory approaches commensurate with the risks posed by the technology.
- Contributing Activity No. 3.7: Support longer-term efforts to develop, as needed, a new non-LWR regulatory framework that is risk-informed, performance-based, and that features staff review efforts commensurate with the demonstrated safety performance of the non-LWR NPP design being considered.

Strategy 4: Facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials)

- Contributing Activity No. 4.1 - Work with stakeholders to determine the currently available codes and standards that are applicable to non-LWRs and their associated fuels and waste, and to identify the technical areas (e.g., instrumentation and control, civil/structural, inservice inspection and testing, materials, equipment qualification, quality assurance, etc.) where gaps exist.
- Contributing Activity No. 4.2 - Participate with the Standards Development Organizations that are actively involved in developing codes and standards for non-LWRs.

- Contributing Activity No. 4.3 - Review codes and standards for endorsement

Strategy 5: Identify and resolve technology-inclusive policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs)

- Contributing Activity No. 5.1: Determine the applicability of previously identified policy issues to non-LWRs.
- Contributing Activity No. 5.2: Identify additional technology-inclusive policy issues for non-LWRs.
- Contributing Activity No. 5.3: Analyze and resolve technology-inclusive non-LWR policy issues identified in Contributing Activity Nos. 1 and 2.

Strategy 6: Develop and implement a structured, integrated strategy to communicate with internal and external stakeholders having interests in non-LWR technologies

- Contributing Activity No. 6.1: Provide timely, clear, and consistent communication of the NRC's non-LWR requirements, guidance, processes, and other regulatory topics, and provide multiple paths for external feedback to the NRC.
- Contributing Activity No. 6.2: Develop consistent NRC non-LWR messaging suitable to a range of audiences.
- Contributing Activity No. 6.3: Promote the exchange of non-LWR technical and regulatory experience with the NRC international counterparts and industry organizations.

3.0 NEAR-TERM TASK PRIORITIZATION & RECOMMENDATIONS FOR USE OF POSSIBLE FY17 OFF-FEE-BASE FUNDS

As noted in the Executive Summary provided in Volume 1,

> Note that the strategies and contributing activities described in this report are assumed not to be constrained by budget or by other agency mission priorities. The purpose of making this foundational assumption is to facilitate the exercise of describing the activities and sequencing needed to accomplish non-LWR readiness, and to estimate the resources that will be needed to complete those activities, without fiscal prejudice. By doing so, the NRC will have in place a work plan that can be executed as resources become available. Resource availability will then govern the pace of achieving readiness, but will not significantly change the activities to be done or the appropriate work sequencing.

For each of the strategy IAPs, the contributing activities and supporting tasks are shown roughly in preferred execution sequence to support the NRC's goal of assuring NRC readiness to effectively, efficiently, and predictably review non-LWRs applications by 2025. This timeframe was selected to align with the Department of Energy (DOE) non-LWR vison and strategy. The NRC recognized that non-LWR vendors may wish to commence pre-application activities or submit applications for review in the near-term, in advance of DOE's deployment goal. In those cases, the NRC will work vendors on design-specific licensing project plans as discussed in strategy 3, and the NRC may accelerate specific contributing activities in this IAP, as needed.

The actual sequencing of the work and actual year of commencement for any specific contributing activity will depend on agency priorities, availability of annual appropriations sufficient to perform the work, and coordination with other NRC organizational initiatives, such as Project Aim.

Given the current non-LWR industry state of technical and regulatory maturity, the staff recommends executing the near-term IAPs within available funding constraints, in an order that first supports ongoing activities:

- Development of the advanced non-LWR design criteria (ARDCs) (technical and regulatory readiness per Strategies 2, and 3)
- Review of near-term regulatory framework flexibilities such as conceptual design assessments and staged-licensing reviews (Strategy 3)
- Facilitation of industry codes and standards development, such as ASME BPV Code, Section III, Division 5 (Strategy 4)

- Continued review and resolution of technology-inclusive policy issues that affect non-LWRs (Strategy 5), and
- NRC non-LWR communications efforts (Strategy 6).

Ongoing activities, such as participation in DOE's Project GAIN, international coordination (e.g., GSAR), OCHCO pilot programs for competency modeling and strategic workforce planning, and continued interactions with DOE's CASL and NEAMS projects should continue as funding permits. Remaining non-LWR research efforts (Strategy 2) and technical readiness activities to prepare the staff to review and regulate non-LWRs (Strategy 1) are also key activities. These efforts should begin as soon as specific non-LWR technology certainty permits.

4.0 NEAR-TERM IMPLEMENTATION ACTION PLANS (IAPs)

4.0.1 General Notes and Assumptions

The IAPs in the following report share a set of common assumptions, listed below. Specific additional assumptions, bases, or other supporting discussions for individual IAPs are included wherever necessary.

- Reference to the document "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness" is made throughout the IAPs. The vision and strategy document is found at Agencywide Documents Access and Management System (ADAMS) Accession No. ML16139A812.
- For the purposes of this report, near-term activities are those performed or initiated during the next five fiscal years (FY17 – FY21). Actual start dates and priorities of the activities shown will be dependent on a range of factors, including NRC work prioritization, actual funding appropriations, industry maturity and application readiness, and similar factors.
- Activities proposed in the IAPs for near-term strategies do not include rulemaking, with the exception of Strategy No. 5 (Policy). Rulemaking will instead be included in the IAPs for the mid- or long-term strategies where required.
- Unless noted otherwise, the near-term activities shown are technology-inclusive.
- Staff assigned to these activities are fully available and qualified when needed. (Assumption does not apply for Strategy 1.)
- Any information needed from industry or other outside organizations (e.g., standards development organizations) is available at the time it is needed.
- The IAPs will be revised when necessary and will benefit from ongoing and future interactions with DOE, industry, reactor designers, and other stakeholders.

4.1 Strategy 1: Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory reviews

Strategy Overview

This strategy supports the NRC's strategic objective of enhancing non-LWR technical readiness. As described in the NRC's vision and strategy for improving the agency's readiness to regulate non-light water reactor (non-LWR) technologies, the strategic objective for enhancing technical readiness is:

> Ensuring that the staff has the requisite knowledge, expertise, tools, and processes needed to efficiently and effectively evaluate non-LWR applications, and to reach an independent safety, security, or environmental finding.

To support accomplishment of this objective, the vision and strategy document described readiness for "people" (the staff) as follows:

> The NRC must have the right number of people with the right skills at the right time in order for the staff to conduct an effective and efficient review. For non-LWRs, the staff must be familiar with a range of potential technologies, must have adequate training support in place, must have a non-LWR knowledge base available, including non-LWR system and integrated plant operations. The staff must also be knowledgeable of any unique waste management, environmental or security challenges posed by a particular non-LWR technology. While many aspects of non-LWR designs may be technology-inclusive (that is, independent of the particular non-LWR technology being reviewed), subject matter expertise for technology-specific aspects of the designs is also required.

The approach taken for this strategy is based on the principle of designing and maintaining the workforce consistent with the work to be accomplished, in the time frame needed. Work design outputs from the contributing activities and support tasks reflected in other near-term IAPs are the drivers for the workforce design, development, and skills maintenance processes.

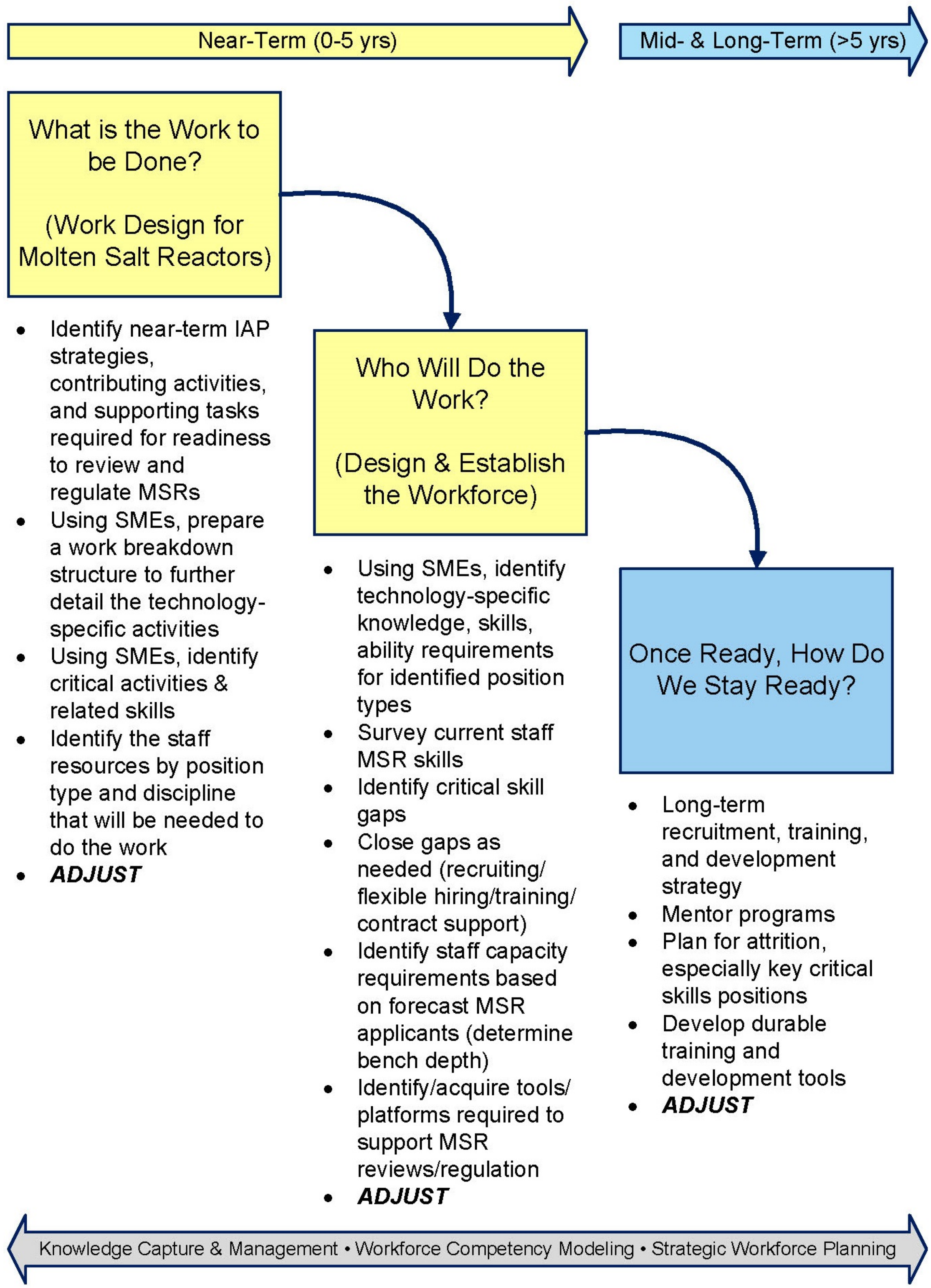
The near-term IAP for this strategy focuses on identification of work requirements, identification of critical skills and staff capacity requirements, assessment of the current staff's non-LWR technical readiness, and technical readiness gap closure by a variety of methods. The mid-term and long-term IAPs will address items such as long-range training and staff development for non-LWRs, mentoring programs, and attrition planning. Certain foundational activities, such as organizational assessments, knowledge capture, knowledge management, workforce competency modeling, and strategic workforce planning are conducted across all readiness preparation timeframes. OCHCO is an integral partner in conducting these foundational activities.

To facilitate the Strategy 1 planning efforts for technology-specific activities, MSRshave been selected as the example non-LWR technology. This technology was selected because, like industry, the staff has the least practical knowledge and experience with MSRs in comparison to the available knowledge base for SFRs and HTGR. Therefore, the preparations and level of effort required to achieve staff technical readiness for MSRs should bound similar readiness efforts for other more familiar non-LWR technologies. Figure 1 illustrates these efforts using an MSR as an example.

The near-term contributing activities and support tasks throughout the IAPs include both technology-inclusive and technology-specific actions. The staff is assumed to be prepared and able to complete the technology-inclusive activities without specialized preparation or training. Technology-specific tasks and the associated critical skills are identified and detailed with the assistance of subject matter experts (SMEs). These SMEs will be identified and sourced from a variety of organizations as needed.

Sources of available non-LWR expertise include the Department of Energy as well as its national laboratories; commercial engineering and regulatory support firms, international regulatory bodies and their research partners; inter-governmental organizations such as the IAEA, NEA, and the Generation IV International Forum; standards development organizations (SDOs) such as ANSI and ASME; and the non-LWR industry itself.

Figure 1 - Strategy No. 1 Overview



Implementation Action Plans – Strategy No. 1

Contributing Activity No. 1.1 - Identify Non-LWR Task and Technical Skill Requirements (Work Design Activities)

The purpose of this activity is to identify the specific tasks that must be performed in the near-term as the NRC prepares to review and regulate MSRs effectively and efficiently (other non-LWR technologies should have similar activities). Using the near-term IAPs as a work planning tool, these tasks will be characterized as technology-inclusive or technology-specific. Technology-specific tasks will be identified and further detailed with the assistance of SMEs. As previously noted, MSRs have been selected as the example non-LWR technology for planning purposes. The information developed for Contributing Activity No. 1.1 will be used as inputs to the workforce training and development activities in Contributing Activity No. 1.2. This work is planned for FY2017 – FY2021.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify the near-term contributing activities and support tasks required to be performed for IAP strategies 2 through 6	X		NRO
Using MSR SMEs, bin the activities into technology-inclusive and technology-specific (MSR) bins (includes staff hours to acquire contract support)	X	X	NRO
Using MSR SMEs, develop further actionable details of the technology-specific readiness activities to be performed	X	X	NRO
Using MSR SMEs, identify the critical activities and related skills	X	X	NRO
Identify the types of staff (by position description and discipline) that will be needed for the technology-specific work.	X	X	NRO

Contributing Activity No. 1.2: Determine and Establish the Necessary Workforce Skills and Capacities (Workforce Design & Establishment)

The purpose of this activity is to perform the near-term activities required to prepare the staff to review and regulate MSRs (other non-LWR technologies should have similar activities). This work shown below is planned for FY2017 – FY2021, but additional related activities may continue into the mid-term (5-10 years).

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify the technology-specific knowledge, skills, and ability (KSA) requirements for the staff identified in Contributing Activity No. 1.1. These are the additional qualifications needed over and above the general KSAs used for LWRs.	X	X	NRO OCHCO
Design and conduct a survey of staff MSR skills and experience.	X		NRO OCHCO
Compare the MSR KSA requirements with the staff survey results and identify critical skill gaps.	X		NRO OCHCO
Prepare a gap closure plan for a minimum set of MSR reviewers. This plan includes identification and description of the required training courses, training system changes (if needed), and any other items required to be ready for training delivery to the staff.	X		NRO OCHCO
Execute the gap closure plan. Develop the required training coursework. Other closure methods may include recruiting staff, using alternative hiring strategies for staff and SMEs, and contracted SME support. This item also includes any software or training tools (not the training courses themselves).	X	X	NRO OCHCO

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Staff training hours – all training methods – minimum set of personnel (Assume 10 critical staff for 720 total hours each, 8 x NRO, 1 x NRR, 1 x NMSS)	X		NRO NRR NMSS

<u>Bases/Assumptions:</u>

1. Current staff with non-LWR experience will be retained and available.
2. Staff qualification can be achieved within 12 months.
3. HRTD has the staff in place to support the development of training delivery or will contract for it accordingly based on the estimates provided in Contributing Activity No. 1.2.
4. Training and development needs for construction inspection (RII) and security (NSIR) will occur in the mid-term timeframe.
5. Research & test reactor development needs will occur in the mid-term timeframe.
6. Identification of further staff capacity needs beyond the initial set of trainees will occur in the mid-term.

4.2 Strategy 2: Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews

Strategy Overview

This strategy supports the NRC's strategic objectives of enhancing non-LWR technical readiness and optimizing regulatory readiness. In support of those objectives, the vision and strategy document states that the staff must have adequate computer models and analytical tools to conduct its review of non-LWR designs in an independent manner.

As part of the staff's review for design certification and licensing of a non-LWR, independent confirmatory calculations of some of the most important design-basis events and key SSCs will be performed. This provides the staff with a basis to examine the applicant's analysis and to confirm the margin of safety for a given design and its operating condition. To perform these independent calculations, the staff will either need to develop or have access to analytical codes suitable for non-LWR application. Currently, the staff has analysis codes that are applicable to conventional and advanced LWRs. For non-LWR reactor designs, the initial tasks will generally include evaluation and down-selecting the codes for use by the staff. This is especially true for design with the least regulatory experience and which have been the subject of only limited code development efforts. The non-LWR technology with the most depth of understanding is the high-temperature gas-cooled reactor (HTGR), resulting from operating experience in the US, UK, Germany, Japan, Russia, and China. Further, in anticipation of gas-cooled reactor licensing in the 2004-2010 time frame in support of the Next Generation Nuclear Plant (NGNP), analytical codes have been selected. SFRs have been constructed and operated in the US, Russia, China, UK, Japan, France and Germany. Of note is France's Rapsodie SFR, which had a particularly long operation period from 1967 to 1983. For molten salt designs, there is far less regulatory review history. An 8 MW thermal molten salt experimental reactor was designed and operated by Oak Ridge National Laboratory (1965-1969).

The approach taken for this strategy is to: 1) identify the computer codes and supporting information and data that would be needed to support both the design of a non-LWR and the staff's review of that design; 2) evaluate the existing computer codes and supporting information to identify gaps in both analytical capabilities and supporting information and data; and 3) interact with both domestic and international organizations working on non-LWR technologies to identify opportunities to collaborate and cooperate in closing the gaps, while being mindful of the importance of avoiding conflicts of interest. The emphasis in the staff's approach is to leverage, to the maximum extent practical, collaboration and cooperation with the domestic and international community interested in non-LWRs with the goal of establishing a set of tools and data that are commonly understood and accepted. The community may comprise NRC, DOE, vendors, utilities, and international regulatory partners. Having a common understanding of the tools and data, rather than having to develop that understanding during each technical review, wcould to significantly improve the efficiency of the review process. NRC can maintain its independence by developing expertise in the codes' phenomenological modeling, numerical schemes, and verification and validation process. NRC will also participate in the development process to the degree that resources allow. It is anticipated that NRC will use the codes to run sensitivity analyses and perform uncertainty analyses to help investigate margins in the design. In some technical areas, an applicant is required to submit the code for NRC's review and approval, such as an evaluation model used for design basis analyses. It would be the applicant's responsibility to justify the quality assurance program used in the code development meets NRC's requirements outlined in Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50. In cases where an applicant uses a code that has been developed by others, commercial grade dedication could be used to verify the quality assurance of the code.

Code development and verification and validation, collectively known as assessment, is extremely resource and time intensive. Therefore, it is not viable for a single organization to undertake all of the required efforts, particularly in light of current budget realities and the deployment timelines that have been suggested by DOE and the industry. Thus, collaboration and cooperation are essential to the success of the strategy.

The staff has a number of ongoing interactions and collaborative efforts with DOE, the domestic research community, and the international community. The approach will build on these existing interactions, developing new cooperative funded activities as appropriate.

For the purpose of developing the IAPs for this strategy, the staff has considered high temperature gas-cooled reactors, sodium-cooled fast reactors, and molten salt reactors where the fuel may or may not be dissolved in the coolant, as the designs of interest in the near-term. This choice is made based on the NRC's experience and is not intended as a "down-select" of the potential non-LWR designs currently being explored by industry and DOE. This design set will be reviewed frequently during the near-term execution of IAP tasks in order to make the most effective possible use of the NRC's resources.

The following sections provide a description of the technologies, and the staff's initial assessment of the current state of the computer codes and supporting information and data. The near-term IAPs involve more structured assessments of the computer codes, information and data, and of the gaps between the current state and what is needed. From those assessments, the staff will further engage the technical community to identify mutual interests and the potential for collaborative and cooperatively funded activities to close the identified gaps.

Based on a preliminary assessment of the gaps, the staff developed a set of contributing activities and general resource estimates for those activities in order to provide a general sense of the efforts and resources that would be needed to close those gaps. This IAP includes a general assessment of the magnitude of the effort that will be required of the non-LWR technical community. This effort will not be funded by NRC alone, therefore, the staff used an approximate value of 25% of the total costs as a rough estimate to inform NRC budgetary needs, as reflected in the tables.

4.2.1 Introduction

There are large number of non-LWR designs that are being considered by industry. Because these designs are still in the conceptual design stage, the proposed research and development is made to be as generic as possible. This can and will be refined as design specifics are made available to the staff, or if particular designs become high priority candidates for design certification and licensing.

As noted in the Strategy Overview, the staff expects to pursue existing interactions with domestic and international organizations, and expand those interactions to close the gaps in computer codes and supporting information and data. In particular for this strategy, the NRC, through its participation, may gain early insights into computer codes and models being developed under the auspices of the GAIN initiative. Continued interactions with DOE's CASL and NEAMS projects will also be beneficial. Both CASL and NEAMS sponsor research and development into advanced analytical codes that may have applicability to non-LWRs. The BISON fuel performance code for example, has the capability to simulate some of the fuel types under consideration for gas-cooled and SFRs. It may be feasible for the community to adopt BISON as its analytical tool, rather than invest in another code such as FRAPCON and FRAPTRAN. Other tools and data being produced by CASL and NEAMS may be of use to the staff in its non-LWR development activities. Thus, continued interaction through joint NRC-DOE briefings as well as collaboration with developers of the CASL and NEAMS tools should be considered.

The NRC could also take advantage of our participation in NEA/CSNI working groups such as WGAMA, WGFS, and GSAR to propose and organize technical research on non-LWRs of mutual benefit to the US and other countries. NEA projects are jointly funded and make use of experimental facilities in member countries, so it may be possible obtain data and initiate experiments in areas where the US may lack facilities. Obtaining agreements with countries with active non-LWR research and development programs could be used to obtain data and operational experience needed by the NRC. Examples include SFR data and operating experience held by Russia, data from the Monju and Joyo facilities of Japan, and gas-cooled reactor data from China and Japan. Specific data and information needs are expected to be produced by the staff during the near-term development period, and a comprehensive list of internationally held data will be identified.

4.2.2 Functional Areas Addressed

In implementing the approach taken for this strategy, as described in the Strategy Overview, the efforts are organized around functional areas. These are: reactor kinetics, fuel performance, thermal-fluids, severe accidents, consequence analysis, and materials and component integrity.

The same analysis tools as used for LWRs can be applied to non-LWRs in the areas of seismic, structural, human reliability, and probabilistic risk assessment (PRA). However, some data would need to be developed in order to complete analyses in these areas. For example, non-LWRs will utilize different components and information on their fragility would need to be developed for seismic response analyses. To support development of a PRA for a non-LWR, modeling of internal and external events will be necessary. For example, SFR designs present unique fire protection considerations and it is likely methods and data would need to be developed; since non-LWRs operating at high temperature may be collocated with chemical facilities to support the use of process heat, new external hazards may need to be considered; and new components will need to be modelled for which failure data may not be readily available.

Non-LWR designs will introduce novel issues in the functional areas of fresh fuel transport and spent fuel storage as well as instrumentation and control. For example, more challenging operating conditions may require the development of advanced instrumentation. These issues will be considered in the mid-term IAPs once designs become more mature and additional information is developed. In addition, issues around fuel storage will also be addressed in the mid-term IAPs.

No functional area is independent, and each depends on others; thus the problems will be multi-disciplinary in scope and effort. Advances in computation that allow a “multi-physics” suite of computer codes to be applied to a problem will help address multi-area issues in an integrated and efficient fashion. This is significant, in that advanced computational capability allows both staff and applicants to more accurately determine safety margins and not be overly conservative.

The following sections discuss the expected research and development in each functional area. The intent is to summarize existing capabilities in each area, and to point out where additional work is necessary based on current understanding of the various non-LWR designs.

4.2.3 Implementation Action Plans

Each functional area described below includes an Implementation Action Plan. These IAPs provide the specific actions to be taken in implementing the approach described in the Strategy Overview. Specifically, to: 1) identify the computer codes and supporting information and data that would be needed to support both the design of a non-LWR and the staff’s review of that design; 2) evaluate the existing computer codes and supporting information to identify gaps in both analytical capabilities and supporting information and data. Once the gaps are identified, specific action plans will be developed to engage the domestic and international community to develop additional collaborative and cooperatively-funded activities to close the gaps.

4.2.3.1 Functional Area: Reactor Kinetics and Criticality

Overview

Currently, the NRC performs independent confirmatory calculations of reactor kinetics and criticality using the Purdue Advanced Reactor Core Simulator (PARCS) (Downar et al., 2006) and Standardized Computer Analyses for Licensing Evaluation (SCALE) (Readren et al., 2014) codes. Based on development of the codes to support future NRC confirmatory analyses of HTGRs during the NGNP program, extensive development work on the codes has been performed and it is likely that the non-LWR technical community may choose these codes as the reactor kinetics and criticality analytical tools to support HTGRs. PARCS has also been developed to support SFR design analysis by NRC’s international regulatory partners. It is not clear if they will be adopted to support analysis for the other designs. As a result, the following sections provide details on what additional development may be needed to support the use of these codes. If these codes are chosen, it is expected that development effort will be supported by the non-LWR technical community. To provide an estimate of budget needs, the NRC contribution to this development has been estimated to be 25%.

To establish the needs of the NRC’s confirmatory analysis capability in the areas of reactor kinetics and criticality safety, a comprehensive functional needs assessment of the SCALE and

PARCS codes must be performed for modeling the proposed non-LWR concepts. This effort should include the following:

- Determination of the functional needs of the codes;
- Determination of conditions and transients to be modeled;
- Determination of the important phenomena that must be modeled through performance of a Phenomena Identification and Ranking Table (PIRT);
- Assessment of the existing reactor core analysis and criticality safety capabilities in the a available tools;
- Identification of phenomenological gaps;
- Identification of data needs to validate the modeling of the important phenomena;
- Collection and organization of available data;
- Development of the codes to simulate the important phenomena;
- Performance of tests needed to obtain the additional data; and
- Validation of the codes with the data.

Important phenomena broadly fall under criticality safety for advanced fuel manufacture, operations, and spent fuel storage and disposal; core physics analysis as it pertains to reactivity control and shutdown margin (steady-state and quasi-steady state analysis), and core physics analysis with respect to coupled fluidic thermal-hydraulic/neutronic transient analysis. Initially, it will be assumed that the Figures of Merit (FOMs) and regulatory acceptance criteria will be defined as part of the policy framework as specific design information becomes available for each reactor type.

For core analysis, engagement with present and past developers of neutronics tools for non-LWRs will be necessary in order to characterize what, if any, adjustments will be needed to the traditional two-step methodology in core physics:

- Cross section generation of isolated bundles at selected branch points;
- Coupled neutronics (nodal diffusion)/fluidic analysis with these cross sections);
- SCALE development areas will include uncertainty assessment for nuclear data libraries, multigroup library generation, and Monte-Carlo reference solution; and
- Verification and validation (V&V), and Monte-Carlo hybrid methods for criticality safety and radiation shielding.

Core physics will be used to demonstrate non-LWR safety via its impact on fuel performance and source term (primary fission product barrier), fluence and its effect on vessel performance (secondary barrier), and as the source of heat transfer through the primary and secondary loops to the associated heat exchangers, turbines, and “containment” (third barrier). Acceptable performance must be demonstrated during normal operating conditions, anticipated operational occurrences (AOOs), design basis transients, and licensing basis accidents.

Criticality safety analysis will be needed to demonstrate safety during fuel manufacture, handling, operation, and intermediate storage and discharge.

Gas-Cooled Reactors

Research and development of neutronics tools for gas-cooled reactor systems were pursued in the 2004-2010 time frame in support of a potential NRC review of the NGNP. Decisions were made to use the PARCS and SCALE codes, and to enhance the capability to simulate single-phase heat transfer to helium coolant. The Advanced Gas Reactor Evaluation (AGREE) (Downar et al., 2015) code was coupled with PARCS as an efficient means of simulating single phase heat transfer in geometries expected for gas-cooled reactors. While work was not completed on PARCS, SCALE and AGREE, the initial efforts for NGNP provide a foundation for a code system appropriate for a variety of gas-cooled reactor designs. Addition effort is expected for:

- Characterization and assessment of the neutron scattering kinematics of thermal systems for graphite reactors (i.e., assessment of S(α,β) scattering laws that have recently been implemented in lattice physics and Monte Carlo codes) (Hawari et al., 2007);
- Enhancement of the current double-heterogeneity modeling tools, including the potential for embedded double-heterogeneity capability (SCALE: Getting Started with SCALE 6.2, 2016);
- Assess the need for higher order methods (transport and/or Monte Carlo) when performing core analysis or cross section generation. For example, if control rods are used for reactivity control (power shaping and zoning) during steady state operation, it may be necessary to adapt or develop new transport methods to characterize control rod worth for shutdown purposes. Transport theory may also be needed for the generation of cross sections along the fuel/reflector interface;
- Vessel fluence calculations will be needed to characterize dose to all of the graphite (fuel and permanent reflector) in order to accurately characterize graphite conductivity (conduction an important heat transfer mechanism during LOFC);
- Vessel fluence will also be needed to calculate “dose” and the consequent embrittlement to the vessel over multiple cycles;
- If U235 enrichment greater than 5% is part of the design, then criticality safety methodologies and benchmarks will need to be adapted and assessed for the proposed configurations during all stages of the cycle (manufacture, operations, and discharge);
- Evaluation of the fine group and broad group library structures that will be used to at the lattice and core levels;
- Development of in-core instrumentation models (neutron detectors and temperature sensors) for core simulators;
- Adaption and implementation of pebble recycle and shuffle algorithm into a core simulator code;
- Modeling of randomly packed graphite pebble fuel forms; and
- Evaluation of the applicability of current methods for calculating tritium production. Tritium is a common by-product in graphite reactors (Massimo, 1976).

There has been significant experience gained through development activities in support of the NGNP. This includes the development of TRIPEN/AGREE for HTGR analysis (OECD-NEA-LOFC) (Takamatsu et al., 2008) and participation in the OECD/NEA MHTGR 350 benchmark (Ortensi et al., 2011). In this benchmark exercise the PARCS codes were exercised by GRS.

In a similar exercise, PARCS (coupled with Thermix) was applied to the OECD-NEA-PBMR-400 benchmark (Seker et al., 2006).

For pebble bed applications, the SCALE code was upgraded, and PEBBED code for pebble movement was developed at Idaho National Lab. SCALE is also applicable to prismatic cores.

SERPENT, a general Monte Carlo transport code that has been tuned for reactors (Leppanen, 2009), has shown promise for accurate cross section generation. For this reason, a new Monte Carlo code (SHIFT) is being developed at ORNL for inclusion as a module within SCALE package (Bowman, 2016).

Sodium-Cooled Fast Reactors

Research on SFRs is currently been conducted within the framework of the Generation IV International Forum, with the participation of several CAMP members. There is also an OECD/NEA benchmark on SFR concepts that will focus on the analysis of the feedback and transient behavior of representative SFR reactor cores (Rimpault, 2016). The SFR benchmark will be purely neutronic, and it will include large and medium size cores, along with several fuel design concepts (metallic, carbide, and oxide fuels).

Sodiumcooled fast reactors have historically been grouped with liquid metal fast breeder reactors (LMFBR), and these could either be characterized as the “pool” type or the “loop” type. Notable designs worldwide have included the EBR-I, EBR-II, and Fast Flux Test Facility (FFTF) within the USA, the Phenix and SuperPhenix test and power reactors in France, and the BN-series test and power reactors within the Russian Federation. In general, SFRs include fuel blocks are shaped into hexagons, with liquid sodium coolant flowing through the blocks. The pressure of the primary system is extremely low in comparison to the primary pressure in saturated water LWR designs. One SFR design that would necessitate NRC anticipatory research is the pool-type, sodium-cooled, breed-and-burn (B&B) fast reactor concept that is being developed by Terrapower, the “TP-1” (Ahlfeld et al., 2011). In this reactor concept, depleted uranium is used to capture neutrons to generate fissile fuel material, and these fuel block types (fertile and fissile) are shuffled to prolong core life and design away the need for re-processing of plutonium. Higher burnup (fissile) assemblies are moved from the inner part of the core to the outer part, with these burned assemblies being replaced by depleted uranium assemblies. The fuel plenum is vented to capture gaseous fission products.

Historically, fast reactor analysis has required a variant of the traditional two-step methodology that has been employed for LWRs. In LWR analysis, more emphasis is placed upon accurate spatial homogenization to capture the physical heterogeneity of the system, with less emphasis on capturing the energy dependence of the flux (two energy group methods are still standard practice at the nodal level). However, in fast-spectrum systems, more energy groups are typically employed at the nodal level to accurately capture the energy component of the flux (Stacey, 2001). That is, fast fissions from the U-235 and Pu-239 nuclides are much more important, and more nodal energy groups are employed to capture the rich resonance structure. As this reactor type depends upon fast-fissions, a moderator is not necessary to slow neutrons down to thermal energies, but higher initial enrichment is initially needed to reach the higher critical mass.

In SFR systems, it is necessary to adjust the parameterization of the reactivity coefficients to account for significant feedback due to fuel displacement in the radial and axial directions, in addition to the Doppler and coolant density feedback. Research in this area will be necessary to ensure that TRACE/PARCS can accurately predict reactor safety significant parameters: fission power and decay heat sources that arise under normal, off-normal, and accident conditions. During a design certification, it would be necessary to independently establish that the reactivity and power can be controlled, and that the shutdown margin is maintained at all points in the cycle. The NRC would also have to confirm that the reactor can be safely shut down without fuel damage during a design basis transient or AOO.

Research of the application of TRACE and PARCS to the analysis of fast spectrum systems has been ongoing within the CAMP community within the past decade. Specifically, the Paul Scherrer Institute (PSI) made adjustments to TRACE/PARCS within the framework of the fast reactor analysis system (FAST), which includes FRED - fuel performance; ERANOS - static neutronics data preparation; TRAC/AAA – system thermal hydraulics; PARCS – multigroup diffusion) (Mikityuk et al., 2005). This capability was developed for the core and safety analysis of critical (and sub-critical) fast spectrum systems, with generic applicability to the several fast reactor types being considered within the Gen-IV International Forum (gas-cooled, lead-cooled, and SFRs).

Fuel transmutation studies with PARCS that converge on a cycle length and nuclide distribution during a search for an equilibrium cycle will be necessary. These studies will be needed to characterize the size of the burning region (fissile) between the spent and fresh fuel regions, and to also calculate the reactivity coefficients (due to Doppler, fuel and structural deformation, etc.) and to confirm that the net reactivity coefficient is negative at all points in the cycle. These studies will also be needed to evaluate the core transient response time (from the smaller delayed neutron fraction that is obtained with bred plutonium during the cycle) and the changes in control rod worth at different points in cycle.

Experimental Needs and Requirements

A thorough review of archived technical reports and operational data that were generated during the operation of the EBR-II (pool) and the Fast Flux Test Facility, FFTF (loop) will be necessary. Potentially, there also may be data collected and published worldwide in the form of critical configurations and start-up tests for fast spectrum systems in Russia, France, Japan, and India. These data sources will be reviewed as a possible validation database.

Molten Salt Reactors

Molten Salt Reactors (MSRs) can be divided into two primary divisions, salt-fueled and salt-cooled, and both of these subclasses have fast and thermal spectrum variants (Holcomb, 2015).

The near-term salt-cooled MSR option is the FHR (Fluoride High Temperature Reactor) that is being studied at ORNL and at several domestic and international universities. The reactor is designed around TRISO particles that are sandwiched into graphite fuel stripes that are affixed into cartridges and loaded into a hex graphite block (Varma et al., 2012). The primary coolant consists of FLiBe ($2LiF-BeF_2$), and the intermediate loop employs Kf-Zr-F4. Many of the physics phenomena that are noted for HTGRs are also evident for the advanced high-temperature reactor (AHTR) as it is primarily graphite-moderated and highly doubly heterogeneous. As in thermal-spectrum HTGRs, tritium production (from tertiary fissions, graphite impurities, and Li-6), transport, and the consequent dose must be considered. As in HTGRs, the TRISO fuel particles are interspersed within a graphite matrix (cartridge). Other HTGR-similar phenomena include the rich resonance structure within the fuel portion of TRISO kernel, the need to consider the fluence-dependent and burnup-dependent graphite scattering kernel, and the random distribution of the fuel grains. Where the AHTR departs from the traditional HTGR is the additional level of heterogeneity from the arrangement of fuel stripes around an interstitial graphite matrix, the need to carefully design around the nuclear properties of FLiBe (moderation, thermalization, and absorption), the unique geometric arrangement of fuel-moderator within a hexagonal structure.

The traditional two-step methodology (pre-computed homogenized cross sections that are fed into a core simulator) may not be applicable, and flux or volume homogenization of lattice data (that preserves reaction rates and multiplication factor) is difficult due to the parallelepiped shape of the fuel cartridges within hexagonal fuel bundles. To date, most of the studies of AHTR have involved either the Reactivity-Equivalent Physical Transformation (RPT) Method (Cisneros and Ilas, 2012) or the Dancoff Correction Method (Kelly and Ilas, 2012). In the RPT Method, TRISO particles are pushed into a smaller active region and then volume homogenization is performed on this smaller, transformed active region. For fuel cycle studies, the volume dimensions are optimized to preserve the beginning of cycle multiplication factor when compared to an explicit, doubly heterogeneous Monte Carlo model of the fuel. In the Dancoff Correction Method, an equivalent Dancoff factor is instead used to force the multiplication factor to factor to match the Monte Carlo reference.

The development of higher order stochastic and deterministic transport methods will be necessary in order to fully capture the multiple heterogeneous nature of the fuel blocks.

These new methods include:

- Multigroup cross sections and the selection of an optimized broad and multigroup structure with consideration for the geometric arrangement, burnable absorbers, control rods, and the energy spectrum including the resonances. This topic includes a characterization for the proper homogenization and dehomogenization of the fuel with consideration for the surrounding regions on the assembly boundary conditions;
- The characterization of the spatial transport mesh within the assembly; and
- The characterization of the scattering kernel within the nuclear graphite.

A review and evaluation of the measurement and processing of nuclear data for graphite and FLiBe moderator/coolant will also be necessary in order to accurately model the fuel. This includes moderation and thermalization of FLiBe, thermalization in carbon, and absorption in carbon.

The most notable example of a fuel-dissolved-in-salt MSR design in the USA is that of the Molten Salt Reactor Experiment (MSRE) at ORNL. This was an extremely successful experiment in which the fuel was dissolved directly in the salt (LiF-BeF_2-ZrF_4-UF_4/ LiF-BeF_2-ZrF_4-UF_4-PuF_3) within a single region core. Currently pursued commercial designs generally requires a once-through design with uranium kept in LEU form, with online gaseous fission product removal and filtering systems being employed. As in HTGRs, tritium production and management is a concern as it is highly diffusive and could potentially be present in the secondary loop.

An accurate methodology for the liquid MSR fuel cycle will be necessary in order to ensure that reactivity control is being maintained within the core and within the fuel startup and feed streams. Given the liquid nature of the fuel, fuel homogenization and energy condensation may not be necessary, and a tailored point kinetics approach may be sufficient to capture the steady state and transient response of the core. This modified point kinetics approach would have specially defined point kinetics coefficients and a delayed neutron fraction that takes into account the flow and movement of fuel, with appropriate scaling for different timescales (a delayed neutron pre-cursor drift model). This methodology would be used for exploratory, scoping studies to bound safety related parameters (net reactivity, values of reactivity components such as Doppler, void, shim (if present), and control rods (if present). This methodology would enable to direct calculation of the reactivity feedback coefficients to ensure that there is no positive net reactivity throughout the cycle. This methodology would be compared to continuous energy Monte Carlo models at a specific point in time.

Experimental Needs and Requirements

The primary experimental data needs for the reactor physics analysis of MSRs mostly consists of the validation of fundamental cross section and lattice data that may be unique to MSRs. This would consist of a review and re-processing of the molten salt (FLiBe and Chloride) and graphite nuclear properties within the AMPX module of SCALE:

- absorption cross sections for graphite and FLiBe;
- characterization of impurities in graphite and the corresponding absorption cross sections; and
- scattering kernel [S(α,β)] kinematics for graphite, FLiBe, and molten chloride.

Ancillary data needs will become evident as more design information becomes available from potential vendors in the future.

References

Taylor, J. M., “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements,” SECY-93-092, 1993.

T. Downar, V. Seker, and A. Ward, "PARCS: Purdue Advanced Reactor Core Simulator," *Proc. Topl. Mtg. Advances in Nuclear Analysis and Simulation (PHYSOR-2006)*, Vancouver, Canada, September 10-14, 2006, American Nuclear Society (2006).

B. T. Rearden, R. A. Lefebvre, J. P. Lefebvre, K. T. Clarno, M. A. Williams, L. M. Petrie, and U. Mertyurek, “Modernization Enhancements in SCALE 6.2”, in *Proc. Topl. Mtg. Advances in Nuclear Analysis and Simulation (PHYSOR-2014)*, Kyoto, Japan, September 28-October 3, 2014, American Nuclear Society (2014).

T. Downar, V. Seker, T. Drzewiecki, B. Kochunas, “AGREE/TRIPEN v1.0: U.S. NRC Core Thermal Fluids and Neutronics Simulator for NGNP. Theory Manual, October 23, 2015. ML15300A234.

Hawari, I.I. Al-Qasir, A. Ougouag, “Investigation of the Impact of Simple Carbon Interstitial Formations on Thermal Neutron Scattering in Graphite.” *Nuclear Science and Engineering* 155, 449-462 (2007).

SCALE: Getting Started with SCALE 6.2, April 2016.

L. Massimo, Physics of High-Temperature Reactors, Pergamon Press, 1976.

K. Takamatsu, S. Nakagawa, T. Takeda, “Core Dynamics Analysis for Reactivity Insertion and Loss of Coolant Flow Tests Using the High Temperature Engineering Test Reactor,” *Journal of Power and Energy Systems*, Vol. 2, No. 2, 2008.

J. Ortensi, M.A. Pope, G. Strydom, R.S. Sen, M.D. DeHart, H. D. Gougar, C. Ellis, A. Baxter, V. Seker, T. J. Downar, K. Vierow, K. Ivanov, “Prismatic Core Coupled Transient Benchmark,” INL/CON-11-20811, 2011.

V. Seker, T.J. Downar, D. E. Carlson, “Analysis of the OECD/NEA/NSC PBMR-400 Benchmark Problem Using PARCS-DIREKT,” Second OECD/NEA/NSC Workshop on the PBMR-400 Coupled Neutronic/Thermal-Hydraulic Transient Benchmark Problem, Paris, France January 26-27, 2006.

J. Leppanen, “PSG2/ Serpant – a Continuous energy Monte Carlo Reactor Physics Burnup Calculation Code Methodology – User’s Manual – Validation Report, VTT Technical Research Centre of Finland, September 5, 2009.

S. M. Bowman, M. Jessee, “SCALE Neutronics Updates for Reactor and Spent Fuel Analysis: Monthly Letter Status Report-June 2016,” NRCHQ6014D0005/NRCHQ6014T0002 (NRC Agreement Number/Task Order Number), Report Period End Date (06/26/2016).
G. Rimpault, “OECD Benchmark for Uncertainty Analysis in Best-Estimate Modeling for Design, Operation, and Safety Analysis of SFRs: SFR-UAM Meeting,” Paul Scherrer Institute, Switzerland, June 2-3, 2016.

Ahlfeld, T. Burke, T. Ellis, P. Hejzlar, K. Weaver, C. Whitmer, J. Gilleland, M. Cohen, B. Johnson, S. Mazurkiewicz, J. McWhirter, A. Odedra, N. Touran, C. Davidson, J. Walter, R. Petroski, G. Zimmerman, T. Weaver, P. Schweiger, and R. Russick, “Conceptual Design of a 500 MWe Traveling Wave Demonstration Reactor Plant,” Paper 11199, *Proceedings of ICAPP 2011*, Nice, France, May 2-5, 2011.

W.M. Stacey, Nuclear Reactor Physics, Wiley-VCH, 1st Edition, 2001.
K. Mikityuk, S. Pelloni, P. Coddington, E. Bubelis, R. Chawla, “FAST: An advanced code system for fast reactor transient analysis,” *Annals of Nuclear Energy* 32 (2005) 1613-1631.

D. Holcomb, “Molten Salt Reactor Fuel Cycle Options,” DOE-NRC Workshop on Advanced Reactor and Accident Tolerant Fuel,” US NRC HQ, Rockville, MD, July 30, 2015.

V. K. Varma, D. E. Holcomb, F. J. Peretz, E. C. Bradley, D. Ilas, A. L, Qualls, N. M. Zaharia, “AHTR Mechanical, Structural, and Neutronic Preconceptual Design,” ORNL/TM-2012/320, September, 2012.

Cisneros, D. Ilas, “Neutronic and Depletion Methods for Parametric Studies of Fluoride-Salt-Cooled High-Temperature Reactors with Slab Fuel Geometry and Multi-Batch Fuel Management Schemes,” *Proc. Topl. Mtg. Advances in Nuclear Analysis and Simulation (PHYSOR 2012)*, Knoxville, Tennessee, USA, April 15-20, 2012.

R. Kelly, D. Ilas, “Verification of a Depletion Method in SCALE for the Advanced High Temperature Reactor,” *Proc. Topl. Mtg. Advances in Nuclear Analysis and Simulation (PHYSOR 2012)*, Knoxville, Tennessee, USA, April 15-20, 2012.

IMPLEMENTATION ACTION PLANS: REACTOR KINETICS AND CRITICALITY

**Develop the Capability to Perform Coupled Nuclear Analysis-Thermal-Fluidic Analysis for High Temperature Gas-Cooled Reactors – SCALE/TRIPEN/AGREE*
[e.g., Confirmatory Codes and Analysis for Design Basis Analysis]**

*this IAP assumes SCALE/PARCS has been selected by the non-LWR technical community.

Contributing Activity No. 2.1: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in the following operating HTGR modes (start-up; quasi-steady state cycle-specific operation; and transient analysis from a limiting point in cycle or equilibrium cycle).

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Finalize SCALE lattice physics capabilities (TRITON) that were developed to analyze double-heterogeneous HTGR fuel (prismatic and pebble) for the NGNP program. • Ascertain source of discrepancy of Assembly discontinuity factors and corner discontinuity factors (ADFs/CDFs) between SCALE and GenPMAXS • Task includes some development, debugging, and testing amongst staff, and support contractors • Expand and document the testing harness (perl) between SCALE and GenPMAXS for hexagonal assemblies and packed pebble assemblies			
FY17	X	X	RES
FY18	X	X	RES
FY19			
FY20			
FY21			
Upgrade the PARCS cylindrical finite difference flux solver into a			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
true cylindrical nodal solver, to support PBR analysis • Involves applied research, as the algorithms have previously been developed amongst various research institutions • Upgrade of PARCS source, the development and revision of cylindrical test problems and test suite, the revision of the documentation, and beta testing.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20			
FY21			
Development of a pebble recirculating algorithm for PARCS • Involves the research, selection, refinement, and implementation of a pebble-recirculation algorithm to complement the cylindrical nodal method developed • Code development, documentation revision(s), test problem development, and beta testing			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Evaluate and develop a methodology to calculate dose (fluence)-dependent scattering kernel in graphite (S(a, b)) for PARCS. [In the current methodology, the graphite			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
scattering kernel from SCALE is considered a static quantity and is not updated with the fluence from the core, but the scattering kernel is heavily a function of fluence]. • Research and evaluate scattering kernel kinematics to determine methods to "update" or "correct" graphite scattering cross section for fluence. This may involve curve fits or extrapolation parameters to "adjust" and/or edit the graphite scattering and transport cross sections from the PMAXS file • This task will involve code development, testing, and documentation.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Evaluate and develop a methodology to calculate fluence and temperature-dependent graphite conductivity with AGREE. Currently, this relies on user input and parameterizations, and transient response during DCC in HTGR depends heavily on the conductivity. This research will also be able to accommodate graphite "shrinkage" with irradiation. • This task will refine an existing AGREE methodology.			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
• This task will involve some code development, testing, and documentation.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Ascertain and evaluate methodology to determine tritium transport through primary and secondary loops and the dose to workers.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES
AGREE Methods Development • Model the cavity cooling system. Secondary system with air or water circulations, coupled with surface radiation and/or natural circulation inside the cavity. • Model air or water into the primary coolant in order to handle air and water ingress accident scenarios			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
AGREE Methods Improvement • Improved radiation heat transfer by adding view			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
factors and allowing node surfaces linked to more than 1 surface. The radiation heat transfer currently considers radiating surfaces are adjacent to each other in the same directions. No multi-dimensional/ multi-surface radiation is currently allowed.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Documentation and test suite. • Theory and users manuals require some improvements. Unit and Regression test suite should be expanded to increase the code coverage to a level consistent with other NRC codes. • Several V&V studies have performed but require documentation. • A formal code performance assessment manual should be prepared.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
The original NGNP plans for TRIPEN/AGREE called for these kernels to be merged into PARCS. TRIPEN was merged into AGREE to form a tightly coupled to code.			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Currently, AGREE does not have an interface to PARCS. Therefore an important next step would be the development of the AGREE interface with PARCS similar to the TRACE and PATHS interface. After the coupling is completed documentation would be performed of the coupled code regression tests. • Merging TRIPEN into PARCS as one of the solution kernels. • Merging AGREE into the PARCS similar to standalone PARCS TH solver and PATHS. • A mapping interface similar to PARCS/PATHs has to be written.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Development of advanced instrumentation to be able to predict power maps and temperature profiles within operating HTGRs. This would rule out any potential "hot-spots" that may be evident in upcoming designs.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Development of PARCS (TRIPEN) in-core detector model for nodal			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
cylindrical and prismatic-hexagonal lattices. Currently PARCS has a detector methodology that was developed for LWR Cartesian nodes. The detector response edit would need to be placed in the SCALE (TRITON) output (t16), and this would need to be read by PARCS (TRIPEN)/GenPMAXS and properly de-homogenized so that the appropriate detector response is calculated.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Assessment of SCALE (KENO-VI) criticality safety capabilities for configurations with greater than 5% enrichment. Enrichments of greater than 5% are being considered for non-LWR fuel designs, the NRC's criticality safety methods will need to be biased for higher enrichments. • Identification of uncertainties in decay chains with higher enrichment. • Revision and/or update of the size of the decay chain (number of nuclides) with enrichment. • Determination of final eigenvalue bias for higher enriched configurations.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.2: Identify experimental data needs and begin code assessment.

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
CRP-5: Resurrect core measurements (temperature, power) and code results that were tabulated within this international program. This program included code-to-code benchmarks developed for the PBMR, GT-MHR, and PBMM; and criticality measurements that were evaluated for the HTR-10 and ASTRA critical facility. • SCALE has already been evaluated against international benchmarks (PROTEUS, HTTR, and HTR-10) developed for the IRPhE Handbook (NUREG/CR-7107)			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21			
OECD/NEA-HTTR-LOFC Program. This is a code-to-date program in which LOFC (Loss of Forced Cooling) occurs when all three Helium blowers (HGC) are tripped. Runs 1-3 at parameterizations of initial power (9MW or 30MW) of VCS activation. Also HTTR start-up criticality and CRW tests have been analyzed with TRIPEN/AGREE. It is expected			

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
that Runs 2 and 3 will be carried out with HTTR restart at the beginning of FY18.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Engage Chinese regulator (NNSA) on collaborations on NRC access to HTR-PM data through the CAMP program. This would include startup physics, ascension testing; operating core data measurements (temperatures and fluxes); and PIE of discharged fuel elements.			
FY17			
FY18	X		RES/OIP
FY19	X	X	RES/OIP
FY20	X	X	RES/OIP
FY21	X	X	RES/OIP
Validation of new coupled PARCS/AGREE methodology and directional diffusion coefficients for TREAT restart program. TREAT is a graphite pulsing reactor that is being restarted. Directional diffusion coefficients are useful for capturing the “neutron streaming” effects inherent in the voided segments of HTGR cores (control rod holes, spaces between pebbles, etc.). Would also help to validate AGREE’s “subchannel” approach to lateral and axial momentum transfer within stacks of graphite blocks	X		
FY17	X	X	RES
FY18	X	X	RES

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
FY19	X	X	RES
FY20			
FY21			

IMPLEMENTATION ACTION PLANS: REACTOR KINETICS AND CRITICALITY

Develop the Capability to Perform Nuclear Analysis for Sodium-Cooled Fast Reactors (SFRs)
[e.g., Confirmatory Codes and Analysis for Design Basis Analysis]

Contributing Activity No. 2.3: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in the following operating SFR modes (start-up; quasi-steady state cycle-specific operation; and transient analysis from a limiting point in cycle or equilibrium cycle)

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop a master fine group library with the AMPX module of SCALE (There are currently 238 and 252 libraries within the SCALE package, but these were developed with thermal spectrums) • Determine the optimum number of fine energy groups (and group structure) to adequately capture all of the resonances, with consideration for the increased importance of fast fissions • Task includes testing and documentation upgrades by support contractor			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	
FY21			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop an optimum broad group energy structure for PARCS. This group structure should be developed with consideration for safety significant core physics parameters (transient power rise, power peakings, control rod worth, shutdown margin, etc.). Previous studies have considered 33 groups. • Task includes testing and documentation upgrades at the PARCS contractor. • Expand and document the testing harness (perl) between SCALE and GenPMAXS for hexagonal assemblies and the additional number of broad groups that are collapsed from SCALE. This will entails the development of additional test problems that are representative of fast spectrum lattices			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X		RES
Perform sensitivity analysis on reactivity feedback components (Doppler, coolant density [sodium void], core radius, core height), the relative importance of each component, and the macroscopic cross section parameterization scheme (macro			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
partial derivative with respect to state parameter). • Task includes testing and documentation upgrades at the PARCS contractor • The development of additional SFR regression and assessment test problems as a result of participation in SFR benchmark exercises. • Upgrading the tentative PARCS assessment manual with SFR analysis			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Research to characterize the relative magnitude of the reactivity components at all points in cycle for a reference SFR design to ensure that the net reactivity coefficient is negative (positive sodium void coefficient). This research will also review postulated transients in the reference SFR design in coordination with the OECD-UAM-SFR benchmark.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.4: Identify experimental data needs and begin code assessment.

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
Participation in the OECD-UAM-SFR Benchmark			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Review archived technical reports, SARs, Tech Specs, and operational data that were generated during the licensing and operation of the EBR-II (pool) and the Fast Flux Test Facility, FFTF (loop). This review would be initiated with the expectation that enough information that would be available to generate SCALE/PARCS/TRACE models of the plant(s). Ideally, RES would have enough information available to converge to a theoretical equilibrium cycle and be able to initiate a postulated AOO from this cycle point (reactivity insertion characterized with a calculated core response).			
FY17	X		RES
FY18	X		RES
FY19	X		RES
FY20			
FY21			

IMPLEMENTATION ACTION PLANS: REACTOR KINETICS AND CRITICALITY

Develop the Capability to Perform Nuclear Analysis-Molten Salt Reactors [e.g., Confirmatory Codes and Analysis for Design Basis Analysis]

Contributing Activity No. 2.5: Upgrade/revise nuclear-analysis capabilities that are capable of predicting core-operating power and flux in an operating MSR, for steady state and transient analysis

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organization
The Transatomic Power design inserts increasing numbers of ZrH moderator rods into the LiF (actinide) F4 molten fuel-salt as the cycle develops. This task can be generalized to other MSR designs without moderator rods. With the design information that is available, it will be necessary to carry out detailed Monte Carlo (SCALE-KENO) calculations to characterize safety significant core physics Figures of Merit. These would need to be parameterized by the fuel-salt displaced volume, point in cycle, feed material (actinide), and the fuel-salt mass flow rate. FOMs would include: • reaction rates (fission, parasitic absorption, Keff, Kinf, power, flux, control rod worth, and moderator rod "worth" throughout the cycle • To determine an optimal group structure for safety analysis, multigroup calculations will need to be compared to continuous energy calculations with KENO • Reactivity components and balance that will vary depending on point in			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organization
cycle (isotopic mix) and spectrum. • Spectrum characterization with point in cycle (balance of thermal to epithermal) • Neutron lifetime and migration length • Reactivity coefficients – fuel-salt void, Doppler, structural expansion, rod worth, fuel-salt mass flow rate, etc.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
For fixed fuel TRISO/FLiBe designs, proper homogenization and energy condensation techniques to enable steady state and transient analysis. This task would define a lattice, define a spatial differencing scheme (point-kinetics, finite difference, nodal diffusion, coupled diffusion-transport, coupled Monte Carlo-transport, or transport), and define an optimal group structure that would accurately characterize an epithermal spectrum			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Ascertain and evaluate methodology to determine tritium transport through primary and			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organization
secondary loops and the dose to workers.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.6: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Through the CAMP program, cooperate with international regulatory partners that are conducting MSR research and constructing test facilities (China, Czech Republic) to obtain relevant data.			
FY17	X		RES
FY18	X		RES
FY19	X		RES
FY20	X		RES
FY21	X		RES
Resurrect data that was collected during the ORNL MSR programs of the 1950s-1970s - Aircraft Reactor Experiment and Molten Salt Reactor Experiment (MSRE) and evaluate as data benchmark			
FY17	X	X	RES
FY18	X	X	RES
FY19			
FY20			
FY21			
Begin code assessment with collected data.			
FY17			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

4.2.3.2 Functional Area: Fuel Performance

Overview

The fuel serves as the first barrier to fission product release for conventional light water reactors. Fuel performance is therefore a critical piece of LWR overall safety performance. Fuel performance for non-LWR reactors must be considered in terms of the safety functions for the fuel as required by the reactor design. Key safety functions such as reactivity control, core coolability, removal of decay heat and fission product retention are demonstrated by a combination of fuel and reactor performance characteristics. Generally, the fuel failure modes and degradation phenomena must be identified for each non-LWR fuel design. Fuel performance principal design criteria (PDC) for each non-LWR design will need to be developed. The NRC recently issued a Solicitation of Public Comments for the Advanced Non-Light Water Reactor Design Criteria (NRC, 2016), which includes proposed criteria related to fuel performance during steady-state and anticipated operations occurrences, criterion 10, and fuel performance in certain accident scenarios, namely criterion 27, 34 and 35. To develop and support fuel-related principle design criteria, the following phenomena will need to be identified for each fuel design:

- Steady-state operation phenomena that alter nuclear, thermal, mechanical, or chemical properties of the fuel;
- Anticipated operational occurrences of the reactor and the fuel response;
- Accident behavior of the fuel at any stage of its life in the reactor;
- Phenomena that will affect the properties of the spent fuel in storage and during normal transport; and
- Accident behavior of the spent fuel for storage and transportation accidents.

Many of the non-LWR concepts utilize fuel with uranium enrichments or plutonium concentrations more than 10 percent. It should be noted that this feature requires careful consideration in the safety analysis of non-LWR fuel manufacturing, operation, and disposal. Existing operating experience for non-LWR reactor technologies provides some information on steady-state operational phenomenon for the associated fuel designs.

Fuel Performance Analysis Codes and Methods

Gas-Cooled Reactors

The non-LWR technology with the most depth of understanding is the high temperature gas-cooled reactor technology, resulting from operating experience in the US, UK, Germany, Japan, Russia, and China. The US efforts associated with the NGNP, efforts in South Africa on HTGR pebble bed designs, efforts in the France on the ANTARES design and efforts in Russia on the GT-MHR design have further contributed to the understanding of HTGR fuel designs. This history allows for reasonable elaboration of fuel performance for HTGRs in the sections below.

The NGNP design proposed tristructural isotropic (TRISO) fuel in their reactor cores. In 2004, NRC conducted a PIRT on TRISO-coated particle fuel for fission product transport due to manufacturing, operations and accidents. The objectives of the TRISO PIRT program were to

(1) identify key attributes of gas-cooled reactor fuel manufacture which may require regulatory oversight, (2) provide a valuable reference for the review of vendor fuel qualification plans, (3) provide insights for developing plans for fuel safety margin testing, (4) assist in defining test data needs for the development of fuel performance and fission product transport models, (5) inform decisions regarding the development of NRC's independent reactor fuel performance code and fission product transport models, (6) support the development of NRC's independent models for source term calculations, and (7) provide insights for the review of vendor fuel safety analyses. The PIRT was published as NUREG/CR-6844 (Boyack et. al., 2004) and won't be reiterated here. In 2010, the NRC issued a High Temperature Gas-Cooled Reactor (HTGR) NRC Research Plan (NRC, 2011). A majority of the information related to fuel performance of HTGR fuel is derived from NUREG/CR-6844 and the Research Plan issued in 2010.

For the NGNP HTGR, the evaluation models for fuel performance and fission product release consist of two distinct components: a stand-alone mechanistic fuel performance analysis tool and the evaluation models or modules for fission product release and transport.

The required models include:

1. Fuel particle failure rate response surface model
2. Core-wide fission product release under normal operation
3. Core-wide fission product release under accident conditions
4. Fission product transport in the reactor coolant system and containment
5. Fuel fabrication and quality assurance inspection methods

DOE/Idaho National Laboratory (INL) has developed a stand-alone fuel performance model, PARFUME. The PARFUME code is designed to predict the behavior of TRISO particle fuel during reactor normal operation and heatup accidents, including the fuel particle failure probability, thereby addressing item (1) above.

The MELCOR code includes preliminary models for core-wide fission product release (items 2 and 3 above). The code also includes fission product transport models for LWR fuel that could be modified, as appropriate, for HTGR fission product transport to address item (4) above. The primary needs in these areas relate to assessment of the preliminary models based on the data being generated in the experimental program on HTGR fuel performance at the Idaho National Laboratory (INL). Based on the assessment outcome, the models will be improved as needed. Work was also done in the past on item (5) at the Oak Ridge National Laboratory. This work will need to be revisited based on the new information obtained from the INL experimental program.

In order to develop the capability to perform fuel performance analysis of high temperature gas-cooled reactors, the NRC would need to (1) develop in-depth knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance, (2) evaluate existing fuel performance analysis codes and determine the feasibility of adopting or updating existing fuel analysis codes applicable to HTGRs, such as the PARFUME code, or determine the need to develop new, independent codes and (3) identify experimental data needs and complete an independent code assessment. Due to the existing understanding of gas-cooled reactor designs, there are no obvious challenges to developing the capability to perform fuel performance analysis for this reactor technology.

The NRC's LWR fuel performance analysis code suite, FRAPCON and FRAPTRAN for steady-state and transient analysis, respectively, are not currently able to analyze HTGR fuel and they would require extensive re-development to achieve this functionality. Most likely, it will be far more practical to iterate an existing code like PARFUME or develop a new fuel performance analysis code from scratch for HTGRs than to modify FRAPCON and FRAPTRAN for HTGR analyses.

Sodium-Cooled Fast Reactors

The majority of the information on fuel performance of ternary fuel clad in HT-9 in SFR designs is replicated from NUREG-1369, NUREG-1368 and NUREG/KM-0007.

The LIFE-METAL computer code is the analytical tool developed at ANL to model the response of metal fuel and blanket elements to steady-state and operational transient conditions. The FPIN2 code is a detailed thermal-mechanical model of an individual fuel element used for analysis of fuel performance under transient conditions. Supporting the FPIN2 code are the STARS code for steady-state initialization and FRAS3 for transient fission gas behavior. The SASSYS code is a whole-core response code that includes a less detailed model for the fuel element thermal-mechanical response.

In order to develop the capability to perform fuel performance analysis of SFRs, the NRC would need to (1) develop in-depth knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance, (2) evaluate existing fuel performance analysis codes and determine the feasibility of adopting or updating existing fuel analysis codes applicable to SFRs, such as the LIFE-METAL code, or determine the need to develop new, independent codes and (3) identify experimental data needs and complete an independent code assessment. Due to the existing understanding of SFR designs, there are no obvious challenges to developing the capability to perform fuel performance analysis for this reactor technology.

The NRC's LWR fuel performance analysis code suite, FRAPCON and FRAPTRAN, are not currently able to analyze SFR fuel and they would require extensive re-development to achieve this functionality. Most likely, it will be far more practical to iterate an existing code like LIFE-METAL or develop a new fuel performance analysis code from scratch for SFRs than to modify FRAPCON and FRAPTRAN for SFR analyses.

Molten Salt Reactors

The NRC's LWR fuel performance analysis code suite, FRAPCON and FRAPTRAN, are not currently able to analyze molten salt reactor fuel and they would require extensive re-development to achieve this functionality. For the molten salt reactors that propose fixed solid fuel (TRISO particles on the inside), an existing fuel code might be applicable. For the molten salt reactors that propose dissolved fuel, NRC is not aware of codes already developed to analyze fuel performance. Developing a code to analyze fuel performance for dissolved fuel will be a multidiscipline effort, likely requiring computational fluid dynamics with coupled kinetics and models to account for changes in fuel composition. In order to develop the capability to perform fuel performance analysis of dissolved fuel molten salt reactors, the NRC would need to (1) develop in-depth knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance, (2) develop a new fuel performance analysis code and (3) identify experimental data needs and complete an independent code assessment. Due to the extremely limited information available to date, it is unlikely that any meaningful work to develop capabilities to perform fuel performance analysis could begin quickly. Within the first 5-years, the staff would develop a code development plan to identify the data needs in more detail and outline interim milestones for this multi-faceted activity.

Experimental Needs and Resource Identification and Development

The experimental needs for developing fuel performance analysis tools vary for each non-LWR reactor design, however it is expected that some amount of new experimental information will be needed in each case. Many of the non-LWR programs that were producing fuel performance data in the 1980's and 1990's are no longer active. The DOE does have advanced fuel research programs ongoing to develop and demonstrate fuel materials, however the lack of experimental facilities with appropriate temperature, flux and coolant conditions makes it difficult to produce enough data to support licensing review. The DOE issued the "Advanced Fuels Campaign (AFC) 2015 Accomplishments" report (DOE, 2015), which includes the latest information on the development of capabilities to support non-LWR fuel development within the DOE complex. The report also includes information on active international collaborations between DOE's AFC researchers and researchers in Korea, France, Japan, China, Russia, EURATOM, and OECD-NEA. The DOE's capabilities and international collaborations go a long way to provide the research resources needed to support non-LWR fuel research. Nevertheless, it would be advantageous to look for additional opportunities to leverage bi-lateral relationships with international regulators in France, Germany, Japan and the United Kingdom, who have historic experience with some of the non-LWR designs being considered.

Gas-Cooled Reactors

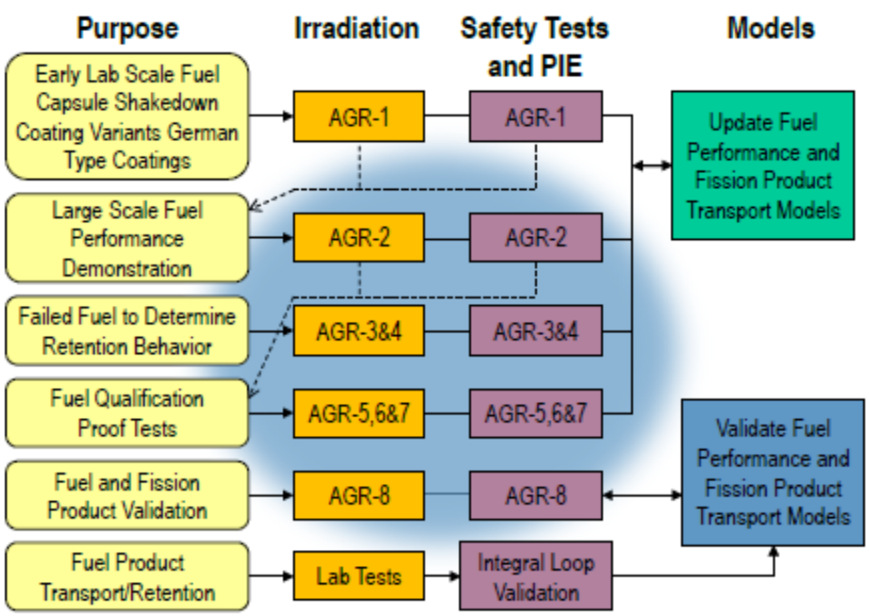
For the NGNP HTGR, the data needs include fuel performance data, fission product release data, fission product transport data and fuel fabrication and quality control data. Regarding fuel performance data, material properties data (e.g., elastic modulus of different coating and buffer layers, SiC strength, PyC anisotropy, etc.) and physico-chemical properties data for unirradiated fuel are needed to support development of the fuel performance model. Similar data for irradiated fuel are also needed for the purpose, in particular, to develop a fuel failure rate response surface model.

Fission product release and transport data needs are particularly important because the HTGR design involves the use of a mechanistic, scenario-specific accident source term rather than a conservative bounding source term. The mechanistic source term must include the transport, retention, and release of fission product s (1) within the fuel element, (2) within structures and surfaces inside the primary pressure boundary, and (3) from the reactor confinement structure. The source term calculation will require a sound technical basis that depends on a sufficient database and modeling of fuel fission product transport and release. Because of the limited operating experience and database for fission product transport, testing of HTGR and VHTR production fuel and fuel materials is needed to develop and benchmark the fission product release and transport models to be used in the mechanistic accident source term calculations over the range of applicable HTGR and VHTR plant operating conditions, transient conditions, and postulated accident conditions. NRC issued an "Assessment of White Paper Submittals on

Fuel Qualification and Mechanistic Source terms (Revision 1) Next Generation Nuclear Plant Project 0748" (NRC, 2014), which provides additional discussion on this topic.

Data on important fuel manufacturing process parameters and fuel product parameters and their associated specifications are needed to develop a guidance document for NRC inspectors. Such data will include fabrication parameters for kernels and coatings as well as matrix and fuel elements, manufacturing process controls and product controls that keep variation of the fuel characteristics within allowable tolerances, and product-sampling analysis methods and data for acceptance verification.

The data to develop a fuel failure rate response surface model are expected to come primarily from the Advanced Gas Reactor (AGR) fuel campaign in the advanced test reactor at INL as part of the DOE-sponsored VHTR R&D. The AGR campaign also will produce fission product release data including metallic and gaseous fission products. An overview of the AGR fuel campaign activities, taken from a 2015 Advanced Gas Reactor Fuels Program Review presentation by David Petti (Petti, 2015), is provided below.



Moisture and air ingress effects are part of AGR-5/6 PIE

The program began in 2002, AGR-1 irradiation and PIE is complete, and AGR-2 and AGR-3/4 irradiations are complete. Fuel fabrication for qualification fuel for AGR-5/6/7 and pre-conceptual design of the AGR-5/6/7 capsule is ongoing. The AGR campaign also will produce fission product transport data through fuel coating and matrix, however data will be needed for transport in the helium pressure boundary and in the confinement. The AGR campaign does not address this need; however, DOE plans to address this need in the future. Test reactors and integral, as well as separate effect test facilities may be used for the purpose. Data needs exist in several areas including dust generation and transport, fission product speciation, plate-out, lift-off, resuspension, and sorptivity in graphite and non-graphite surfaces.

Sodium-Cooled Fast Reactors

The NRC reviewed two SFR designs in the 1980's and 1990's, SAFR and PRISM, that both utilized metallic ternary fuel comprised of uranium, plutonium and zirconium with a HT-9 steel cladding. This fuel system has little operational experience. The EBR-II reactor in Idaho had many years of successful operation with metal fuel, however the differences in material, geometry and operating conditions between the EBR-II and the SAFR and PRISM reactors make it difficult to apply that experience without additional fuel and material testing, safety tests, and analytical model development.

In the Preliminary Safety Evaluation Report for the PRISM and SAFR designs, the staff cited the following data needs to support the establishment of fuel design limits and fuel damage limits for licensing, and for the validation of the analytical tools for licensing evaluations:

- the uniformity of quality (for example, the composition, thermo-physical properties, and strength characteristics) resulting from production and fabrication technologies for the fuel and cladding;
- behavior and extent of fuel restructuring and porosity characteristics as a function of burnup (> 10 at. %) and the development of Zr-depleted regions and potential plutonium distribution; and the axial strain limits;
- fuel-cladding eutectic formation temperature; cladding wastage, or penetration rate, as a function of temperature; cladding failure mechanism(s); and run-beyond-cladding-breach data;
- data for fast, rapid reactivity insertion, transients to quantify the axial extrusion reactivity feedback to establish the energetics of a hypothetical core disruptive accident, and the behavior of molten fuel during a power excursion;
- the statistical data base to support the claim of <0.01 percent fuel failures;
- the conclusions drawn with respect to the behavior of the PRISM fuel system under "slow" overpower transients to be verified in experiments with fuel elements of prototypic geometry; and
- source term data including fission-product release from fuel matrix, the transport and holdup in the sodium pool, the transport and holdup in the cover gas region above the sodium pool, and the transport and holdup within the containment boundary

The DOE programs cited in the 1990's to supply data for the SFR designs included EBR-II, Transient Reactor Test Facility (TREAT) and FFTF. EBR-II and FFTF have since shut down. The TREAT reactor was shut down, but may be restarted. The lack of U.S. experimental facilities with a fast flux spectrum presents a challenge to the materials and fuel performance behavior research that will be needed to qualify SFR fuel designs. The DOE previously collaborated with researchers at the Phenix fast reactor in France to conduct FUTURIX-FTA experiments to address this facility gap.

In addition to the US experience, it is worth noting that both France and Japan have operated large scale SFRs used for commercial power generation, namely Super-Phenix and Monju, respectively. In both cases, MOX fuel was used, and significant experience and data may be available from these commercial reactor projects.

Molten Salt Reactors

For molten salt designs, there is far less information on fuel performance, which makes it difficult to identify the experimental needs and requirements. Much of the interest today in reviving the MSR concept relates to using thorium (to breed fissile uranium-233). There are a number of different MSR design concepts, and a number of interesting challenges in the commercialization of many, especially with thorium. Some of the MSR designs burn plutonium or depleted uranium. Molten salt designs utilize fuel without cladding and therefore fission product release must be mitigated using non-cladding design features. Control of tritium would be a key factor.

Again, the lack of U.S. experimental facilities with a fast flux spectrum presents a challenge to the materials and fuel performance behavior research that will be needed to qualify molten salt reactor fuel designs.

References

NRC, "Solicitation of Public Comments for the Advanced Non-Light Water Reactor Design Criteria," April 7, 2016, Accession Number: ML16096A420

Boyack B. E., Morris R. N., Petti D., Powers D. A., "NUREG/CR-6844, Vol 1, TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations and Accidents," July 2004, Accession Number: ML042300668

NRC, "High Temperature Gas-Cooled Reactor (HTGR) NRC Research Plan," January 2011, Accession Number: ML103560222

King T. L., Landry R. R., Throm E. D., Wilson J. N., "NUREG-1369, Pre-application Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid Metal Reactor," December 1991, Accession Number: ML063410547.

Carlson D. E., "NUREG-1368, Pre-application Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," February 1994, Accession Number: ML063410561

Flanagan G. F., Madni I. K., Mays G. T., "NUREG/KM-0007, NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors," April 2014, Accession Number: ML14128A346

DOE, "Advanced Fuels Campaign (AFC) 2015 Accomplishments," February 2015, Available on the web at: https://nuclearfuel.inl.gov/

NRC, "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source terms (Revision 1) Next Generation Nuclear Plant Project 0748," July 2014, Accession Number: ML14174A845

Petti, D., "2015 Advanced Gas Reactor Fuels Program Review," Available on the web at: https://art.inl.gov/ART Fuels Program Review/03 Petti AGR Fuels Program Review.pdf

IMPLEMENTATION ACTION PLANS: FUEL PERFORMANCE

Develop the Capability to Perform Fuel Performance Analysis of High Temperature Gas-Cooled Reactors

Contributing Activity No. 2.7: Develop knowledge of fuel design, fuel functional requirements, and fuel characteristics critical to safety and accident performance.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop knowledge of the fuel behavior under irradiation (e.g. fission product retention/release, SiC/ZrC failure mechanisms) and its interaction with the coolant. Identify existing regulatory considerations.			
FY17			
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21	X		RES/NRO
Identify or confirm conceptual Anticipated Operational Occurrence, Design Basis Accident, and Beyond Design Basis Accident space.			
FY17			
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21	X		RES/NRO
Develop knowledge of the front- and back-end fuel cycle strategy for HTGR fuel (including enrichment levels). Identify key regulatory considerations.			
FY17			
FY18	X		RES/NRO/NMSS
FY19	X		RES/NRO/NMSS
FY20	X	X	RES/NRO/NMSS/
FY21			
Develop knowledge of existing fuel manufacturing requirements and quality control points. Identify key regulatory considerations.			
FY17			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY18			
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21			

Contributing Activity No. 2.8: Develop or adopt/update existing fuel analysis code applicable to HTGRs.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Assess existing PIRTs (NUREG/CR-6844 and 6944) and update as needed for additional HTGR types and/or fuel designs.			
FY17			
FY18			
FY19	X	X	RES/NRO
FY20			
FY21			
Review existing TRISO fuel performance codes and evaluate need for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X		RES
Identify potential shortcomings in the fuel performance code and formulate a plan for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X		RES
FY19	X		RES

FY20	X		RES
FY21	X		RES
Perform code and model development. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.9: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify existing operational and experimental databases used previously for code assessment and determine limits of applicability (e.g. burnup, power) of existing data. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES
Identify missing experimental information necessary for code assessment based on proposed fuel design and operating conditions. Obtain data from on-going applicant-sponsored research as made available. This will be done collaboratively with			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20			
FY21	X	X	RES
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20			
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: FUEL PERFORMANCE

Develop the Capability to Perform Fuel Performance Analysis of Sodium-Cooled Fast Reactors

Contributing Activity No. 2.10: Develop knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop knowledge of fuel design for each SFR design. Identify key regulatory considerations.			
FY17			
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21	X		RES/NRO

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop knowledge of the relationship of fuel performance to thermal-fluid and plant performance (including fission product and tritium retention/release - will fuel design provide fission product barrier function?). Identify key regulatory considerations.			
FY17			
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21	X		RES/NRO
Identify or confirm conceptual Anticipated Operational Occurrence, Design Basis Accident and Beyond Design Basis Accident space			
FY17			
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/NRO
FY21	X		RES/NRO
Develop knowledge of the front- and back-end fuel cycle strategy for SFR fuel (including enrichment levels). Identify key regulatory considerations.			
FY17			
FY18	X		RES/NRO/NMSS
FY19	X		RES/NRO/NMSS
FY20	X	X	RES/NRO/NMSS
FY21			
Develop knowledge of the fuel manufacturing requirements and quality control points. Identify key regulatory considerations.			
FY17			
FY18			
FY19	X		RES/NRO/NMSS
FY20	X	X	RES/NRO/NMSS
FY21	X	X	RES/NRO/NMSS

Contributing Activity No. 2.11: Develop or adopt/update existing fuel analysis code applicable to SFRs

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for SFRs by general type; pool and loop. This work will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES/NRO
FY20			
FY21			
Review the available fuel performance codes and evaluate for further development. This work will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X		RES
Identify potential shortcomings in the fuel performance code and formulate a plan for further development. This work will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X		RES

FY19	X		RES
FY20	X		RES
FY21	X		RES
Perform code and model development. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.12: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data from on-going applicant-sponsored research as made available. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20			
FY21	X	X	RES
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20			
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: FUEL PERFORMANCE

Develop the Capability to Perform Fuel Performance Analysis of Molten-Salt Reactors

Contributing Activity No. 2.13: Develop knowledge of fuel design, fuel functional requirements and fuel characteristics critical to safety and accident performance.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop knowledge of fuel design for each MSR design. Identify key regulatory considerations.			
FY17			
FY18			
FY19			
FY20	X		RES/NRO
FY21	X		RES/NRO
Develop knowledge of the relationship of fuel performance to thermal-fluid and plant performance (including fission product and tritium retention/release - will fuel design provide fission product barrier function?). Identify key regulatory considerations.			
FY17			
FY18			
FY19			
FY20	X		RES/NRO
FY21	X		RES/NRO
Identify or confirm conceptual Anticipated Operational Occurrence, Design Basis Accident and Beyond Design Basis Accident space.			
FY17			
FY18			
FY19			
FY20	X		RES/NRO
FY21	X		RES/NRO
Develop knowledge of the front- and back-end fuel cycle strategy for MSR fuel (including enrichment levels). Identify key regulatory considerations.			
FY17			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY18			
FY19			
FY20	X		RES/NRO
FY21	X		RES/NRO
Develop knowledge of the fuel manufacturing requirements and quality control points. Identify key regulatory considerations.			
FY17			
FY18			
FY19			
FY20			
FY21	X		RES/NRO

Contributing Activity No. 2.14: Develop fuel analysis code applicable to MSRs.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for MSRs by general type; pool and loop. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES/NRO
FY20			
FY21			
Review the available fuel performance codes and evaluate for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify potential shortcomings in the fuel performance code and formulate a plan for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20	X		RES
FY21	X		RES
Perform code and model development. This will be done collaboratively with the non-LWR technical community. Based on current knowledge of existing codes, resources are estimated. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20			
FY21	X	X	RES

4.2.3.3 Functional Area: Thermal-Fluid Phenomena

Overview

The term thermal-fluids phenomena refers to the physical processes involved in normal operation and design-basis accidents. The focus is on the fluids involved in the reactor coolant system, and well as those in the intermediate heat exchange systems through the ultimate heat sinks. The primary goal of thermal-fluid analysis is analysis of the safety systems necessary to remove decay heat following an incident. In general, the analysis of thermal-fluid phenomena is performed for design-basis events; loss-of-coolant accidents, loss-of-flow, control rod ejection, etc., such that the core geometry remains intact and there is limited damage to the fuel. Note that “thermal-fluids” is termed rather than thermal-hydraulics, as there is a wide variety of fluids proposed for use in non-LWR designs.

The coolant used in the primary system is significant in that its selection can either simplify or complicate the analysis. Wateris somewhat unique and complex in that most systems allow it to boil and two-phase flow becomes an important part of the evaluation. It often leads to large uncertainties due to the nature of a two-phase flow. Single-phase fluids, such as helium-cooled systems or in other designs where boiling is highly unlikely, are generally easier to analyze. Computational fluid dynamic (CFD) is often a reliable means of evaluation for single phase fluids, even with complex geometries. Molten salt reactor systems may be complicated by conditions in which solidification can occur or when conditions are attained in which thermal-fluid properties are less certain.

The following sub-sections discuss the challenges and research and development efforts associated with thermal-fluids phenomena for advanced non-LWR reactors. The safety issues with thermal-fluids phenomena are often multi-disciplinary in nature. Significant interaction with fuel, materials, and neutronics feedback is expected in non-LWRs. Thus, coordination between thermal-fluid phenomena research and other areas should be anticipated.

Analysis codes and Evaluation Model requirements are highly dependent on the coolant, reactor system design, and the associated safety systems both active and passive. This subsection briefly describes reactor design considerations, along with the accident scenarios and phenomena that will be of primary interest. To identify code requirements and phenomena of interest, the staff expects to develop a PIRT for each design as specific design information becomes available. At the current time, PIRTs are not available for SFR or MSR designs, but under the NGNP program PIRTs were developed for HTGR for both prismatic and pebble bed designs. For the SFR and MSR designs, the scenarios and phenomena of interest considered here are only preliminary.

Gas Cooled Reactors

Two types of gas-cooled reactor designs, modular HTGRs with pebble bed cores and modular HTGRs with prismatic block cores, may be submitted for design certification. Both types use helium as the coolant and use a Brayton cycle to obtain high thermal efficiencies. Plants with both direct cycle and indirect cycle (i.e., with an intermediate heat exchanger (IHX) between the reactor and balance of plant) are being considered. Because of previous work to support the NGNP, the thermal-fluid infrastructure has been under development by the NRC.

While the full spectrum of accident scenarios considered as part of an HTGR design basis has not yet been firmly established, existing studies of PBMR and GT-MHR have shown that a loss of normal heat removal is an important type of accident to be modeled for assessment of design margins. For dose consequences, events involving the loss of pressure boundary are generally the most severe. Accidents in the former category include the loss of forced circulation (LOFC) with reliance on passive heat removal systems as an important accident scenario. If the system pressure boundary remains intact, the reactor pressure is maintained and the event is called a pressurized LOFC transient or a "pressurized cooldown." The coolant is not lost during this type of event, and the helium coolant remains at high pressure. Heat is removed by radiation from the core to the reactor pressure vessel wall, and then through successful operation of the passive reactor cavity cooling system (RCCS). Buoyancy and natural convection circulation play an important role in the core and reactor pressure vessel temperature distributions, with the chimney effects tending to make temperatures highest near the top of the reactor pressure vessel for the pressurized conduction cooldown. Thus, the primary thermal-fluid analysis needs are to determine the core temperature distribution and maximum fuel temperature, while the design criteria of interest internal to the core relate to the fuel, and the design criteria external to the core relate to maximum temperatures for the vessel and support system components.

Phenomena and features that are expected to be important considerations in infrastructure development for gas-cooled reactors include the following:

<u>Buoyancy-Driven Ingress</u>. In depressurized LOFC events with air or water ingress, molecular diffusion was originally considered to be the most important process that initially transports air or water vapor into the lower plenum. Since the rate of diffusion of air through helium is slow, several hours may pass before enough oxygen could be transported through the lower plenum to the lower core reflector or fuel at the bottom of the core so that significant oxidation could occur. Information obtained since development of the thermal-fluids PIRT has indicated however that buoyancy driven flow through an opening in the reactor vessel can cause ingress of air or water vapor at a much faster rate and cause oxidation to begin much earlier in a depressurized LOFC event (the process is sometimes referred to as "lock-exchange"). In a buoyancy-driven exchange, the relatively high density mixture of air and helium in the containment drives flow into the vessel replacing the low density helium that counter flows into the containment. Thus, in a depressurized LOFC event, buoyancy-driven ingress is important in initiating oxidation in the lower plenum and core.

<u>Natural circulation and buoyancy</u>. Natural convection flow and heat transfer are important in several events. In the pressurized LOFC event, natural convection acts to make temperatures in the core and vessel relatively uniform. Rising hot plumes of helium entering the upper plenum may cause the upper reactor vessel head temperature to rise to unacceptably high temperatures. In the depressurized LOFC event with air or water ingress, natural circulation becomes important in heating or cooling the core following the start of oxidation.

<u>Graphite oxidation</u>. The oxidation of lower reflector graphite and fuel near the bottom of the active core can occur during depressurized events with air ingress. The heat of reaction enhances circulation within the core and vessel. Availability of air is important in the evaluation, as there may not be enough oxygen to sustain significant oxidation rates since the reactor cavity

is well below grade. In addition, the heat release from graphite oxidation is approximately equal to decay heat and thus represents a significant contributor to heatup of the fuel.

Pressure drop through a pebble bed core. Simulation of the flow resistance through a pebble bed core is important in obtaining the temperature distribution in the fuel. Bypass around the pebbles near the central and outer reflectors will depend on the resistance through the pebble bed and wall drag along the reflector flow path. The simulation may need to account for random variations in the packing fraction and their impact on the flow.

Core heat transfer. Heat removal from the core to the reactor vessel wall depends on several individual heat transfer mechanisms. In a pebble bed core, heat is transferred by conduction and radiation from pebble to pebble and from pebbles to graphite structures. Convection to the coolant occurs, but it is not as effective as conduction and radiation without forced circulation. In depressurized LOFC events, the effective core conductivity is the dominant parameter and uncertainty for heat removal from the core. This effective conductivity depends on the relative contributions of conduction heat transfer through the core, reflector, and vessel structures.

“Graphite dust” transport. During normal operation, particles of graphite can become dislodged from fuel due to abrasion, wear and aging. Fission products can diffuse into these graphite particles. This so-called “graphite dust”, especially for pebble bed reactors, can therefore become an important if not dominant contributor to fission product transport from the fuel particles to the reactor coolant system internal surfaces and to the environment during a helium pressure boundary break. There is a need to develop data and models for the transport of this graphite dust within and out of the reactor coolant system. In particular, there is a need to develop data and models for the prediction of the remobilization, transport and release of the accumulated graphite from the reactor coolant system and confinement building. Release of graphite dust into the reactor cavity and surrounding building raises the possibility of ignition/explosion of the suspension of graphite dust in air and air-helium mixtures.

Reactor cavity heat transfer. Similar to core heat transfer, reactor cavity cooling is dominated by radiation heat transfer. The RCCS in both the pebble bed and prismatic block reactors designs relies on passive natural circulation (through vents) and radiation heat transfer to remove heat from the reactor pressure vessel. In effect, the RCCS acts as the link between the reactor pressure vessel and the ultimate heat sink (e.g., the ground). Some RCCS designs are water cooled and, while these designs are passive in nature as the water circulation is driven by natural circulation, they do contain active components. Specifically, water chillers are used to maintain subcooling so that evaporation is avoided. The assumed failure of these active components would then cause the water cooled RCCS to transition from a single-phase natural circulation mode to a boildown of the liquid inventory. Thus, a two-phase flow model of the RCCS may be necessary.

Reactivity insertions. Two types of events have the potential to cause reactivity insertions resulting in recriticality. Depressurized LOFC events with water ingress may transport sufficient water to the core to cause a significant reactivity insertion. Pressurized LOFC events with anticipated transient without scram (ATWS) can also achieve recriticality later in the event after xenon decay. Simulation of this process depends not only on the core neutronics but also on the temperature and water vapor distribution in the core.

If the non-LWR technical community were to select NRC codes for use as the analytical tool to perform thermal-fluid analyses for HTGRs, additional code developmental work would be needed. To provide accurate core operating temperatures and power distributions and to analyze reactivity-initiated transients, such as water ingress events or ATWS, the systems analysis code would need to be coupled with the PARCS-AGREE code. A CFD code, such as FLUENT or STAR-CCM+, will be used to provide detailed local temperature and velocity distributions, especially for relatively open regions such as the plena and the RCCS. A reactor systems analysis codes, such as MELCOR, can provide the overall, global systems behavior of the HTGR and be coupled to fuel failure and fission product transport models. For all code types -- systems, core or CFD, significant code development and assessment remains necessary to improve and quantify models and correlations for gas reactor phenomena. Details on these codes follow.

The combined PARCS-AGREE code system allows for detailed 3-D calculations of the power and temperature distributions in a gas-cooled core as was demonstrated in the PBMR-400 benchmark calculations. PARCS-AGREE will need to be modified for applicability to the pebble bed design. The Advanced Gas Reactor Evaluation (AGREE) module was added to the PARCS code to handle the gas dynamics and heat transfer processes for a pebble bed reactor. AGREE is a 3-D equivalent of the well-known 2-D THERMIX-DIREKT code.

Several organizations have used FLUENT and other CFD codes to examine the details of flows in AGRs. The ability of these codes to simulate turbulent mixing in complex geometries makes them well suited for analysis of flows in the upper and lower plena of HTGRs where buoyant plumes and hot jets in the lower plenum may exist. Natural convection flow and heat transfer dominate cooling in the RCCS, and CFD may be needed to effectively examine the details involved in operation of gas reactor systems. The staff may also need a CFD capability to calculate the steady-state distribution of radionuclides on the internal surfaces of the pressure boundary system to provide the initial conditions for the calculation of the initial releases in the source term analysis.

MELCOR is a severe accident code developed at Sandia National Laboratory for the NRC to model the progression of accidents in light-water-cooled reactors. MELCOR models have been developed to simulate most aspects of a pebble bed reactor. Modifications include implementation of multi-fluid tracking capabilities, a graphite oxidation model, and a simple molecular diffusion model. Correlations were also added to model heat transport in a pebble bed and to include the effect of neutron fast fluence on thermal conductivity of the core. A study conducted by INL used MELCOR, which allows for general and flexible nodalization, to develop a detailed model of the reactor pressure vessel and RCCS.

Another code, Graphite Reactor Severe Accident Code (GRSAC), developed at ORNL, can simulate a wide range of accidents in gas reactors. GRSAC has been used to simulate both the prismatic and pebble bed reactor designs, as well as benchmark transients run in the HTTR and HTR-10 integral test facilities. The forerunners of GRSAC, called ORECA and MORECA, were developed in the 1975 to 1993 time-frame at ORNL to support the staff's licensing safety evaluation for Fort Saint Vrain and the pre-application review for the DOE modular high-temperature gas-cooled reactor (MHTGR). GRSAC provides for a one-dimensional flow solution with a detailed three-dimensional conduction model of the core, plus models for the reactor vessel, shutdown cooling system, and RCCS.

Assuming PARCS-AGREE are selected as the thermal-fluid and kinetics codes for HTGRs by the non-LWR technical community, near-term tasks and objectives are:

- Identify and select HTGR accident scenarios;
- Use the HTGR PIRT to identify potential shortcomings in the thermal-fluid code and formulate a plan for further development;
- Identify experimental information necessary for code assessment;
- Obtain applicable experimental data for code assessment;
- Continue development of PARCS-AGREE to simulate the high ranked phenomena identified in the PIRT; and
- Initiate code assessment.

The following work will be completed in the mid-term time frame. Mid-term tasks and objectives for HTGRs are:

- Complete code development to add or improve thermal-fluid modeling capabilities;
- Perform experimental programs to produce additional data, as needed; and
- Finalize code assessment on a representative design.

As part of the long-term efforts once designs mature, the tasks and objectives are:

- Perform code assessment and validation specific to a gas-cooled reactor design; and
- Perform studies to quantify uncertainties for design basis accident scenarios.

Sodium-Cooled Fast Reactors

SFRs use liquid sodium as the primary reactor system coolant, which allows for high power densities and efficient heat transfer while the coolant is a single phase. The primary system operates at near atmospheric pressure (0.1 to 0.5 MPa) with normal maximum outlet temperatures of approximately 550 C, which is roughly 400 C below the boiling point.

Of particular interest and concern in the safety of SFRs, is prediction of the occurrence of coolant boiling. This can result from an uncontrolled loss of flow leading to a positive reactivity feedback. The subsequent power excursion can initiate a core disruptive accident. Local flow blockages can cause cladding dryout and damage to the fuel assembly. Thus, a precise prediction of the flow and temperature distribution within the core and its subassemblies is necessary under both normal and accident conditions.

An additional concern is thermal oscillations and the fatigue of structures subjected to the oscillating stresses. Once again, a detailed understanding of the flow and temperature distribution in the core and reactor vessel is required.

During 2009-10, the NRC participated in a CSNI sponsored group called TAREF (Task group on Advanced Reactor Experimental Facilities) to provide identify on LMR test facilities in the US and elsewhere. The report (TAREF, 2010) on SFRs included information on metallic fuels and summarized safety issues. The TAREF used a process similar to PIRT to identify the main technical issues. Safety related issues the group concluded would need additional study for SFR thermal-fluid phenomena included:

- Flow regime transitions
- Transport properties
- Channel flow distributions
- Sodium boiling
- Coolant-structure interaction
- Natural convection

As SFR designs mature and information is developed, accident scenarios and safety-related thermal-fluid phenomena will be thoroughly identified. This will likely involve development of a PIRT for a specific design and consideration of the design's safety systems.

A primary near-term task will be for the non-LWR technical community to evaluate the needs for its confirmatory analytical capability for SFR safety analysis. The community will evaluate thermal-fluid systems codes that can potentially perform safety analysis for the SFRs under consideration. Selection of a code or codes for thermal-fluid analysis will likely be preceded by a PIRT for SFRs. Selection of a code or code suite for staff use will pave the way for development of additional analytical capability, if found necessary based on applicability requirements dictated by the PIRT. The following provides information on codes currently available for SFR systems analysis and the analysis of thermal-fluid phenomena.

SASSYS-1 (Cahalan, 2007) was developed by Argonne National Laboratory (ANL) for the Department of Energy (DOE) in the mid-1980s. SASSYS-1 is designed to perform deterministic analysis of DBAs and BDBAs in liquid metal cooled reactor (LMR) plants. Detailed, mechanistic models of steady-state and transient thermal, hydraulic, neutronic, and mechanical phenomena are employed to describe the response of the reactor core, the reactor primary and secondary coolant loops, the reactor control and protection systems, and the balance-of-plant to accidents caused by loss of coolant flow, loss of heat rejection, or reactivity insertion. The consequences of single and double-fault accidents are modeled, including fuel and coolant heating, fuel and cladding mechanical behavior, core reactivity feedbacks, coolant loops performance including natural circulation, and decay heat removal. SASSYS-1 analysis is terminated upon demonstration of reactor and plant shutdown to permanently coolable conditions, or upon violation of design basis margins. The objective of SASSYS-1 analysis is to quantify accident consequences as measured by the transient behavior of system performance parameters, such as fuel and cladding temperatures, reactivity, and cladding strain. Originally developed for analysis of SFRs with oxide fuel clad by stainless steel, the models in SASSYS-1 were subsequently extended and specialized to metallic fuel clad with advanced alloys. The models in SASSYS-1 have been validated with extensive analyses of reactor and plant test data from EBR-II and FFTF.

The SSC-L and SSC-P codes were originally developed at Brookhaven National Lab for the NRC in 1978 and 1980, respectively. SSC-L (Agrawal, 1978) is a version designed to analyze loop-type SFRs while SSC-P (Madni and Cazzoli, 1980) is a version designed to analyze pool-type LMRs. Both codes calculate the thermal-fluid response of SFR systems during operational, incidental, and accidental transients, especially natural circulation events. Modules simulated and parameters calculated include: core flow rates and temperatures, loop flow rates and temperatures, pump performance, and heat exchanger operation. Additionally, SSC-L and SSC-P account for all plant protection and plant control systems. Although the primary emphasis is on transients for safety analysis, these codes can be used for many other applications, such as scoping analysis for plant design and specification of various components. Any number of user-specified loops, pipes, and nodes are permitted. Both single- and two-phase thermal-hydraulics are used in a multi-channel core representation. Inter-assembly flow redistribution is accounted for using a detailed fuel pin model. The heat transport system geometry is user-specified. The codes provide steady-state and transient options and a restart capability. A number of natural circulation tests have been performed in the FFTF (loop-type) and the PHENIX and EBR-II (pool-type) reactors. Predictions from SSC-L agree with the FFTF results and predictions from SSC-P agree with the PHENIX and EBR-II results.

The SSC-K code (Chang, et al., 2002) was developed by the Korea Atomic Energy Research Institute (KAERI) to analyze the Korea Advanced Liquid Metal Reactor (KALIMER). SSC-K is a modified and updated version of SSC-L to perform analysis of pool-type reactors such as KALIMER. SSC-K includes a two-dimensional pool model for analysis of the thermal stratification phenomena in the hot pool and a sodium boiling model for the core.

CERES (Nishi et al., 2006) is a transient plant system analysis code for LMRs developed by the Central Research Institute of Electric Power Industry (CRIEPI) and the Japan Atomic Energy Agency (JAEA). CERES analyzes the primary, secondary and auxiliary cooling system, the thermal-hydraulic characteristics in the reactor vessel plenum and the flow characteristics inside the intermediate heat exchanger. The reactor plena is modeled in 2D or 3D. Plant components such as pipes, pumps, steam generators, etc. are modeled in one-dimension. CERES uses a one-point kinetic model and 2-D (R-Z) reactivity feedbacks. Analytical results from CERES have shown good agreement with plant trip test at MONJU. CERES has been used in the design of the 4S reactor.

ARGO-3 (Horie, et al., 2008) is a plant dynamics code for SFRs developed by the Toshiba Corporation. ARGO-3 is used to estimate the safety performance of the fuel pins and the primary coolant boundary under DBEs and their state under BDBEs. ARGO-3 has been used to perform safety margin calculations for the 4S reactor design.

EPRI-CURI-L, P (Khatib-Rahbar and Cady, 1981) is a code used to analyze operational transients, anticipated incidents and postulated accidents that do not lead to sodium boiling in loop-type (L) and pool-type (P) LMRs. EPRI-CURL was developed at Cornell University in 1979 for the Electric Power Research Institute (EPRI). Models include point reactor kinetics, primary, intermediate and tertiary system heat flow, coolant flow dynamics governed by forced and natural convection effects and plant protection and control systems. EPRI-CURL appears to have not been applied to a reactor design (I. K. Madni, 2010).

The TRAC and RELAP Advanced Computational Engine (TRACE) code developed by the NRC has some capability for SFR analysis. TRACE has been updated to simulate liquid metal coolants as part of a project to simulate accelerator transmutation of waste. The update added liquid metal properties for sodium and lead-bismuth, a fluid conduction model, and liquid metal heat transfer correlations. The liquid metal models were limited to single phase flow. Several members of a TRACE User's Group have investigated using TRACE for liquid metal cooled reactors. Paul Scherrer Institute (PSI) added sodium boiling and multi-phase flow models to TRACE and now uses TRACE as part of their a system to analyze fast spectrum reactors. Assessment of TRACE applicable to SFRs includes work by Chenu et al. (2009) and by Tenchine et al. (2013) for the Phenix reactor. TRACE has also recently been used to model a BN600 type SFR (Zhang and Mikityuk, 2016).

Near-term tasks and objectives for SFRs for thermal-fluid phenomena and development of tools for systems analysis are:

- Develop PIRT(s) for SFRs by design type -- pool and loop;
- Identify and select SFR accident scenarios;
- Use the PIRT to identify potential shortcomings in the codes and select a code;
- Formulate a plan for further development;
- Identify experimental information necessary for code assessment;
- Obtain available applicable experimental data for code assessment;
- Perform code development to add or improve thermal-fluid modeling capabilities; and
- Initiate code assessment.

The following work will be completed in the mid-term time frame. Mid-term tasks and objectives for SFRs are:

- Integrate, or couple with other codes (fuel performance, neutronics, etc.) necessary to perform accident simulations;
- Perform experimental programs to produce additional data, as needed; and
- Finalize code assessment using separate effects and integral effects data on a representative design.

As part of the long-term efforts once designs mature, the tasks and objectives are:

- Perform code assessment, development and validation specific to an SFR design; and
- Perform studies to quantify uncertainties for design basis accident scenarios.

Molten-Salt Reactors

Molten salt reactor designs use either a molten fluoride or molten chloride salt as the coolant in which the fuel can be either solid or dissolved in the coolant itself. The molten salt coolant is usually a binary or ternary eutectic mixture in order to lower the melting point. Molten salts have very high boiling points (>1200 C). Dissolved fuel MSRs enable on-line refueling and removal of fission products. Because of the high heat capacity and high boiling point of molten salt coolants, boiling of the coolant is unlikely and MSRs thus have features that make the design passively safe. Because molten salts have not received considerable attention, some properties such as viscosity and specific heat, may need better quantification at high temperatures.

Loss of forced flow accidents are likely to represent safety significant scenarios, as they reduce heat transfer to the fuel, and can cause heat up. Scenarios that involve blockages to coolant flow (perhaps due to localized freezing of the salt) may also be a concern. Safety significant accident scenarios and phenomena of interest will need to be identified as part of future work.

There has been limited development of codes and evaluation models that are applicable to molten salt reactors (MSRs). There has however been some work performed to develop and apply TRACE to a conceptual molten salt reactor design (Wang, et al. 2015). TRACE has been updated to include the properties of several salt coolants (Richard et al. 2014), which allowed the code to be used for exploratory studies. In the initial work, the fuel was assumed to be a solid carbon-carbon composite retained in plates. Additional work however will be necessary to implement models and correlations appropriate for the specific fuel design selected if TRACE were to be selected.

Significant efforts should be expected in development of thermal-fluid codes suitable for molten salt reactors in which the fuel is in solution and transported by the coolant. Development of an analytical capability for a molten salt reactor containing liquid fuel will likely require coupling between a thermal-fluids code and a neutronics code in order to simultaneously simulate the power, temperature, and velocity profile within the fuel/coolant mixture. Gao et al. (2013) have performed such an analysis for a Gen-IV molten salt reactor by coupling a multiple-channel analysis code (MAC) and MCNP. The non-LWR technical community can consider this approach, possibly by coupling a CFD code with neutronics code.

Near-term tasks and objectives for MSRs for thermal-fluid phenomena and development of tools for systems analysis are:

- Develop PIRT(s) for MSRs by design type -- pool and loop;
- Identify and select MSR accident scenarios;
- Use the PIRT to identify potential shortcomings in the codes and select a code;
- Formulate a plan for further development;
- Identify experimental information necessary for code assessment;
- Obtain available applicable experimental data for code assessment;
- Perform code development to add or improve thermal-fluid modeling capabilities; and
- Initiate code assessment.

The following work will be completed in the mid-term time frame. Mid-term tasks and objectives for MSRs are:

- Integrate, or couple with other codes (fuel performance, neutronics, etc.) necessary to perform accident simulations;
- Perform experimental programs to produce additional data, as needed; and
- Finalize code assessment using separate effects and integral effects data on a representative design.

As part of the long-term efforts once designs mature, the tasks and objectives are:

- Perform code assessment, development and validation specific to a MSR design; and
- Perform studies to quantify uncertainties for design basis accident scenarios.

Experimental Needs and Requirements

Code development is highly dependent on verification and validation, collectively known as assessment. Codes must be thoroughly assessed against applicable data in order to assure that predictions of a hypothetical accident scenario are accurate for a full-scale plant. Because experimental data at full-reactor design scale is generally not available, code developers must utilize two type of tests; separate effects tests thatexamine specific phenomena, and integral effects tests thatprovide information on system performance. Both type of tests must appropriately scaled to conditions expected in the full-scale plant, both geometrically and in the thermal-fluids range of conditions. Thus, access to well-scaled, applicable experimental data is central to the code development and assessment process.

For non-LWRs, part of the code development effort will require the collection and evaluation of test data for specific designs. While there are significant amounts of data for non-LWRs, it is not immediately clear if these data are sufficient or applicable to future designs. Thus, near-term efforts will focus on this collection and evaluation of the data. Augmenting this effort will be a monitoring of DOE and applicant test programs. Many of these data will be beneficial and possibly essential to the non-LWR technical community's code development program.

It is very possible that there will be "holes" in the existing database, and new tests and new test facilities will be needed. Thus, another part of the initial efforts will be to identify shortfalls in the database and to conduct some experimental programs either to provide data for development of models or to confirm the safety margins in a design. As a result, the mid- and long term efforts involve use of DOE/applicant test facilities and data.

References

Taylor, J. M., "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements," SECY-93-092, 1993.

Nuclear Energy Agency, "Experimental Facilities for Sodium Fast Reactor Safety Studies, Task Group on Advanced Reactor Experimental Facilities (TAREF)" NEA/CSNI/R (2010)12, Paris, France, 2011.

J.E. Cahalan, "Computer Codes: SASSYS-1, Standard Code Description," ANL Website, July 17, 2007. < http://www.ne.anl.gov/codes/sassys/index.html>

A.K. Agrawal, *Advanced thermohydraulic simulation code for transients in LMFBRs (SSC-L code)*, BNL-NUREG-50773, February 1978.

I.K. Madni and E.G. Cazzoli, *Advanced thermohydraulic simulation code for pool-type LMFBRs (SSC-P code)*, NUREG/CR-1738, September 1980.

W.P. Chang, Y.M. Kwon, Y.B. Lee and D. Hahn, "Model development for analysis of the Korea advanced liquid metal reactor," *Nuclear Engineering & Design*, 217, 63-80 (2002).

Y. Nishi, N. Ueda, I. Kinoshita, A. Miyakawa and M. Kato, "Verification of the plant dynamics analytical code CERES using the results of the plant trip test of the prototype fast breeder reactor MONJU," *International conference on nuclear engineering (ICONE 14)*, Miami, FL, July 17-20 (2006).

H. Horie, K. Miyagi, K. Nakahara and H. Matsumiya, "Safety performance of the 4S reactor on the ATWS events – statistical estimation of uncertainty," *Progress in Nuclear Energy*, 50, 179-184 (2008).

M-Khatib Rahbar and K.B. Cady, *EPRI-CURL-P Systems Modeling for a Pool-type Liquid Metal Fast Breeder Reactor*, Final Report, EPRI-NP-2170, December 1981.

Chenu, K. Mikityuk, and R. Chawla, NS&D, V 239 (2009) 2417-2429.

D. Tenchine, D. Pialla, T.H. Fanning, J.W. Thomas, P. Chellapandi, Y. Shvetsov, L. Maas, H.-Y. Jeong, K. Mikityuk, A. Chenu, H. Mochizuki, S. Monti , "International benchmark on the natural convection test in Phenix reactor", NE&D 258 (2013) 189-198.

Y. Zhang and K. Mikityuk, "Static and transient analysis of a medium-sized sodium cooled fast reactor loaded with oxide, nitride, carbide and metallic fuels", Annals of Nuclear Energy, 87 (2016) 761-771.

Dean Wang, Graydon L. Yoder, David W. Pointer, and David E. Holcomb, "Thermal hydraulics analysis of the Advanced High Temperature Reactor," Nuclear Engineering and Design, 294 (2015) 73–85.

Joshua Richard, Dean Wang, Graydon Yoder, Juan Carbajo, David Williams, Benoit Forget, Charles Forsberg, "Implementation of Liquid Salt Working Fluids Into TRACE," Proceedings of ICAPP 2014 Charlotte, USA, April 6-9, 2014, Paper 14214.

Madni, I. K., "Survey of Safety Analysis Codes for Liquid Metal Cooled Reactors (LMRs)," Draft Report of the U.S. Nuclear Regulatory Commission, August 2010.

Zhangpeng Guo, Jianjun Zhou, Dalin Zhang, Khurrum Saleem Chaudri, Wenxi Tian, Guanghui Su, Suizheng Qiu, "Coupled neutronics/thermal-hydraulics for analysis of molten salt reactor," Nuclear Engineering and Design 258 (2013), 144–156.

IMPLEMENTATION ACTION PLANS: THERMAL-FLUID PHENOMENA

Develop the Capability to Perform Thermal-Fluid Analysis of Gas-Cooled Reactors [e.g., Confirmatory Codes and Analysis for Design Basis Analysis]

Contributing Activity No. 2.15: Develop thermal-fluid analysis code applicable to gas-cooled reactors.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Continue development of PARCS-AGREE, and with increased emphasis on efforts for simulation of pebble bed reactors. This IAP assumes PARCS/AGREE is selected by the non-LWR technical community to serve as the thermal-fluid analysis tool. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X	X	RES
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.16: Identify experimental data needs and begin code assessment.

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment, and begin perform assessment against available data. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X		RES
FY18	X		RES

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: THERMAL-FLUID PHENOMENA

Develop the Capability to Perform Thermal-Fluid Analysis of Sodium-Cooled Fast Reactors [e.g., Confirmatory Codes and Analysis for Design Basis Analysis]

Contributing Activity No. 2.17: Develop thermal-fluid analysis code applicable to SFRs.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for SFRs by general type; pool and loop. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X	X	RES
FY18			
FY19			
FY20			
FY21			
Review the available thermal-fluid codes and select an analysis tool. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
necessary resources to complete this work.			
FY17	X		RES
FY18	X	X	RES
FY19			
FY20			
FY21			
Identify potential shortcomings in the thermal-fluid code and formulate a plan for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20			
FY21			
Perform code and model development. This will be done collaboratively with the non-LWR technical community. The resources shown represent NRC work.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.18: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data as made available. This will be done collaboratively with the non-LWR technical community. The resources shown			

represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: THERMAL-FLUID PHENOMENA

Develop the Capability to Perform Thermal-Fluid Analysis of Molten Salt Reactors [e.g., Confirmatory Codes and Analysis for Design Basis Analysis]

Contributing Activity No. 2.19: Develop thermal-fluid analysis code applicable to molten salt reactors

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for MSRs by general type; fixed-fuel and circulating fuel. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X	X	RES
FY18			
FY19			
FY20			
FY21			
Review the available and potential thermal-fluid codes and methods,			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
and select a code. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X	X	RES
FY19			
FY20			
FY21			
Identify model development and experimental information needs. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20			
FY21			
Perform code and model development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.20: Identify experimental data needs and begin code assessment.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data as made available. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.		0	
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

4.2.3.4 Functional Area: Severe-Accident Phenomena

Overview

Offsite consequence analysis is the final aspect of a PRA, the so-called Level 3 for LWRs. The mix of radionuclides and the chemical forms in the releases from accidents, including all levels of severity, occurring in non-LWRs might differ from those in releases during accidents in LWRs. Therefore, evaluations of non-LWR technologies and comparisons with existing LWRs will require the comparison of the off-site consequence level.

Severe Accident Analysis and Source Term Codes and Methods

The NRC's LWR severe accident codes, based on many experiments performed in the 1980s following the Three Mile Island 2 accident, include MELCOR, SCDAP/RELAP5, CONTAIN, VICTORIA, and IFCI. As NRC's consolidated LWR severe accident code, MELCOR can model most aspects of a severe accident including thermal-fluid analysis, core melt progression, fission product release, and transport in the reactor system, and containment. For LWRs, the United States and other nations have performed many experiments to develop a fundamental understanding of the phenomena of severe accident and fission product transport. The recent NRC focus on severe accidents has included upgrading MELCOR and benchmarking it against the more specialized severe accident codes (e.g., SCDAP/RELAP5 and VICTORIA) and experimental results. The TEXAS code is used to analyze fuel coolant interaction phenomena. NRC has not modified MELCOR to simulate severe accident behavior in non-LWRs. The following sections provide a summary of severe accident phenomena likely to be important in system response and fission product release. For each of the design types, a PIRT will need to be developed to finalize the identification and ranking of the phenomena. This information will then be used to help select a severe accident code to support analyses for each of the design types, to identify code development needs to model the important phenomena, and help define the experimental programs to develop adequate data for model development and code validation.

Gas Cooled Reactors

For assessment of dose consequences from accidents in gas-cooled reactors, the scenarios of greatest interest for in-core and ex-core temperature calculations involve the failure of the helium pressure boundary with the attendant LOFC. These events may range from a small leak to the postulated double-ended guillotine break of the cross-connection inlet/outlet pipes (or vessels) that are used to transfer helium between the reactor unit and the intermediate heat exchanger (IHX) or directly to the power conversion system. In loss of helium pressure boundary (HPB) scenarios, the reactor system eventually depressurizes to atmospheric pressure. For the large breaks in the DBE or beyond-DBE category, decay heat is removed from the core by conduction, convection, and radiation, and from the reactor vessel wall to the RCCS by radiation and convection. For small breaks in the AOO category, decay heat may be removed initially by the decay heat removal system and subsequently by the RCCS. Large HPB breaks involving RCCS cooling are also referred to as "depressurized conduction cool down events." Thermal convection effects for helium at atmospheric pressure tend to be insignificant for this type of event.

During normal operation, some amount of "graphite dust" will be generated due to vibration and wear between moving fuel and graphite blocks that are in contact with the fuel pebbles. The rate of generation of this graphite dust is expected to be greater in pebble bed reactor cores than in prismatic cores. Information from the AVR test reactor indicated that graphite dust generation in a pebble bed core could be significant. Graphite dust will contain fission products (e.g., cesium), and thus for HPB breaks that attain velocities sufficient to mobilize and transport graphite dust, an increase can occur in dose consequences. If graphite dust is found to be readily transported and contains sufficient fission products, it could be a major contributor to the initial fission product release and, consequently, an increase in dose can occur at the site boundary and beyond prior to significant fuel particle failures due to accident core heatup. Accurate calculations of the three-dimensional distribution of core temperatures during normal operation are important for determining the level of fission products in the graphite dust.

The release of graphite dust into the reactor cavity and surrounding building creates the possibility of a dust ignition and/or rapid burning. The ignition and rapid burning limits of graphite dust suspensions in air and air/helium mixtures should be determined.

Convection cooling and natural circulation within the vessel are important in depressurized LOFC events involving the ingress of air into the HPB vessel system. Depending on the HPB break or location of the opening into the system, relatively high-density air either diffuses or flows via the lock-exchange mechanism into the vessel. Lock exchange refers to the counter-current stratified flow of two fluids with significantly different densities. The oxygen in the air oxidizes in the graphite core supports, reflectors, and active fuel region. Heat generated by oxidation of the hot graphite can result in higher rates of circulation within the vessel and convective heating or cooling of the core. The amount of air (i.e., oxygen) available limits the oxidation in the active core region, and the graphite oxidized maybe restricted to that in the lower core supports and bottom reflector and bottom of the active fuel region.

Steam ingress can be a concern in depressurized LOFC events if the system design includes water-cooled heat exchangers or intercoolers. Even though the water-cooled system components operate at a pressure significantly less than the helium coolant, if the boundary between the two systems should fail, the water-cooled system could over-pressurize, and water could enter the reactor system via gravity flow. Although transport of water through the primary and into the core appears to be a very low probability event, water ingress into the core could also result in a reactivity increase and possible recriticality. In addition, water ingress is of concern because of the potential damage to coated fuel particle (CFP) s due to chemical attack and increased fission product release from failed coated fuel particles.

HTGR events that lead to fission product release from the reactor pressure boundary must be modeled. Source term (i.e., fission product release and transport) analysis methods will be needed to estimate the magnitude, composition, and timing of fission product release from the reactor pressure boundary, from the containment or confinement system, and to the environment.

HTGR DBE and BDBE analysis and source-term analysis also will be needed to support the development of limiting sequences. Therefore, fission product release and transport data and accident progression analysis codes and the expertise to apply them will be needed both to estimate consequences of events that contribute significantly to the integrated plant risk and to evaluate specific HTGR safety and technical issues.

For HTGRs, and other non-LWRs that significantly differ from operating and advanced LWRs, event sequences and the fission product transport and release process (e.g., graphite dust release from a pebble bed reactor pressure boundary) can also differ. In HTGRs, fission products may be released from intact CFP defects and heavy metal contamination from manufacture, diffusion during normal operation and accidents, release from CFP failures during normal operations and accidents, and by lift-off of the fission product plated out on cool metallic surfaces during normal operation.

The risk from fission product releases from an HTGR pressure boundary may be associated with AOOs, DBEs, and BDBEs. The fission product release from the fuel for these event categories can occur as a result both of diffusion through and release from failed CFPs. Technical expertise in the area of fuel fission product transport and behavior during normal operation and licensing-basis events is needed to assess HTGR event consequences and overall plant risk. Because fission products released from the fuel are transported through the primary system and containment predominantly as aerosols, the offsite releases and offsite radiological consequences may be significantly reduced by fission product deposition on surfaces within the pressure boundary and on surfaces within the containment or confinement system. Aerosol deposition occurs through a variety of mechanisms, such as gravitational settling, thermophoresis, and diffusiophoresis. Graphite dust, especially for a pebble bed reactor design, may contain a significant fraction of the fission products released during normal operation and due to its small size (approximately 1 micron) can also be modeled as an aerosol. The remobilization of deposited graphite dust during a depressurization event can then greatly enhance the initial release source term. As graphite dust behaves differently than the plate-out for condensable fission products, models for the deposition and resuspension of graphite dust will be required. Data and modeling on fission product release and transport resulting from the effects of fuel oxidation and convective flow within the core and out of the helium pressure boundary will be needed.

MELCOR has most of the capabilities needed to analyze beyond-DBA accident issues. Modifications include implementation of multi-fluid tracking capabilities, a graphite oxidation model, air ingress model, and a simple molecular diffusion model. Correlations also were added to model heat transport in a pebble bed reactor and to include the effect of neutron fast fluence on reducing graphite thermal conductivity of the core. A pebble bed accident analysis sensitivity study conducted by INL used MELCOR. MELCOR allows for general and flexible nodalization to develop a detailed model of the reactor pressure vessel and RCCS.

However, if MELCOR is selected by the non-LWR technical community to serve as the severe accident analysis tool for HTGRs, it would need to be modified because of the differences between LWR and HTGR designs. These include differences in the fuel, core, and reactor internal structure design and differences in the material properties for the fuel, core, and core support structures and coolant. To support code modification needs assessment, a PIRT was conducted to delineate the important HTGR accident phenomena and factors, including fission product release and transport phenomena to be modeled by the accident analysis code, as well as the experimental data required for model development and assessment. Planned initial modifications identified are described below:

Extend fission product release models. Release models in the code will need to be expanded to capture current fission release models, which are based on Core Source Term Release (CORSOR), CORSOR-M, or Booth formulation to predict release from HTGR fuel (e.g., spherical fuel pebbles, prismatic block fuel). The modifications will include the effects of fuel and core temperature, air or steam oxidation, and burnup on fission product release and transport. The fission product release models need to include release from kernel, retention in matrix material, diffusion through iPyC and OPyC layers, transport through structural materials, and release into confinement and to the environment including the effect of filtered venting.

Expand oxidation models. The current oxidation models for various materials in the code will need to include a graphite oxidation model. Oxidants to be considered for the model should include oxygen, steam, and moist air. The oxidation model should account for CO and CO_2, as well as H_2 in the case of steam oxidation, where CO may further react with O_2. The oxidation model should be able to predict self-sustaining oxidation (i.e., graphite burning). Oxidation models for pebble bed reactor fuel (i.e., graphite matrix material) and core structural, moderator, and prismatic fuel blocks (i.e., nuclear-grade graphite material) will be needed. In addition, the model should consider smoke and particulate formation.

Update materials properties models. Fuel and structural material components in MELCOR must include graphite. Key graphite fuel matrix and metallic component properties include conductivity, emissivity, specific heat capacity, and diffusion coefficients. The models should consider graphite/fuel degradation and relocation, as well as the strength and integrity of core supporting structures. The core modeling capabilities should allow for both pebble bed and prismatic designs. The models should be able to calculate core effective thermal conductivity including the effect of gray gas on radiation heat transfer. Graphite behavior modeling should include dimensional changes (swelling and shrinkage) in graphite due to irradiation, and the effect of dimensional changes on emissivity, matrix retention, bypass flows, etc.

Reactor Cavity Cooling System. The reactor cavity cooling system (RCCS) is a critical safety-related SSC for proposed HTGR designs. Two RCCS designs have been proposed: air-cooled (for the GT-MHR) and water-cooled (for the PBMR). For the air-cooled design, the coolant flow inside the RCCS panels is driven by natural convection. Whereas, for the water-cooled design, both natural convection and forced flow designs have been considered. Both of the water-cooled designs employ a chiller located in a tank external to the reactor cavity to remove the decay heat. The operation of this chiller is dependent upon electrical power, so a loss-of-power event will eliminate the chiller as a heat sink requiring the decay heat to be removed by evaporation of the water inventory which is the basis for design basis accident DBA core cooling. Consequently, the model for the water-cooled design has to be able to accommodate two-phase flow within the RCCS. In addition to handling the air or water flows inside the RCCS, the code will need to calculate with reasonable accuracy the combined natural convection and radiation heat transfer from the reactor vessel wall to the RCCS. Moreover, in the event that a partial failure of the RCCS is postulated coincident with a depressurized loss-of-forced-convection cool down—perhaps due to the fluid-structure interaction during the blowdown—the code will need to be able to consider the effect of non-axisymmetric heat transfer on the vessel exterior.

Graphite dust models. As discussed above, a significant fraction of the fission products released during normal operation may reside in graphite dust deposited within the pressure boundary during normal operations. Models for the deposition and resuspension of this dust will need to be incorporated into MELCOR and assessed.

Improve numerics. MELCOR numerics will need to use longer time steps to carry out reasonable execution times for slowly developing accidents. This may involve changing the numeric solver for MELCOR to implement the semi-implicit two-step algorithm.

Evaluate the need for additional fission product deposition/transport experiments and HTGR models. Once model implementation in the MELCOR code is complete, there will be a need to evaluate the code against available integral fission product transport experiments. There is also a need to model selected non-LWR designs and demonstrate code capabilities for selected scenarios.

To achieve this objective, a literature review is needed to identify HTGR experiments on fission product release during high-temperature transport and deposition conditions within the reactor pressure boundary and containment or confinement under accident conditions. Because the release of non- fission product FP aerosols from the core increases fission product aerosol deposition, this literature review should include experiments on aerosol releases of other core materials under accident conditions. Based on the results of the literature review, there will be a need to assess the need for additional experiments.

The results of the above R&D will be a version of the MELCOR integrated severe accident code capable of analyzing the progression of AOOs, DBEs, and BDBEs (including severe accidents) in either pebble bed reactors or prismatic designs. This version of MELCOR could be used with input from other codes that establish initial conditions and boundary conditions (e.g., initial fission product distribution within the helium pressure boundary surfaces, initial failed versus intact coated particles) to determine source term and fission product release.

Sodium-Cooled Fast Reactors

SFR have been studied, designed, and operated since the 1950s. They have the ability to breed new fissile material from more-common fertile isotopes, and hence utilize a significant fraction of the mined uranium (or thorium). SFRs have high power densities and heat removal requirements, requiring very low neutron moderation and necessitating the use of a liquid metal coolant, such as sodium. Power variations due to neutron leakage at the core boundaries resulted in the need for ducted assemblies and control of coolant flow.

Accident categories for SFR are typically in one of three categories, loss of cooling (unprotected loss of flow, ULOF), loss of normal heat removal (unprotected loss of heat sink, ULOHS), and reactivity addition (unprotected transient overpower, UTOP). During operation of some designs, the fuel is not in the most reactive condition in the core – relocation of fuel, particularly oxide fuel, during an accident would increase reactivity, possibly exceeding prompt critical conditions. These accident categories, particularly the ULOF, were the focus of licensing discussions for the proposed Clinch River Breeder Reactor, and at the time that the project was terminated, these issues were not completely resolved.

After the Clinch River Breeder Reactor project was canceled in the early 1980s, attention turned to studying and developing more advanced fast reactors that would also be “inherently safe.” These new designs promoted passive safety features that were based on fundamental physical processes such as natural circulation, thermal expansion, gravity, and lower fuel melting temperatures with consequent fuel dispersal into a less-reactive state (inside cladding that has a much higher melting temperature).

The materials present in an SFR core include liquid sodium, fuel, and cladding. Under normal operating conditions and all DBA conditions the sodium is in liquid state, while the fuel and cladding are in solid state. For the representative SFR, a postulated LOF accident with failure to scram in an oxide-fueled SFR will lead to coolant boiling, associated core voiding, and fuel melting and relocation. The most important reactivity feedbacks due to material phase changes are those associated with the sodium boiling and fuel melting and relocating. The onset of sodium boiling, if it occurs, will tend to increase the core reactivity and power potentially leading to fuel melting and dispersal. The melting and relocation of the fuel has a complex effect on the core reactivity, because the molten fuel can relocate both inside the fuel pin cladding and outside, in the coolant channel.

As the power increases, the inside of the fuel pin begins to melt, forming an internal cavity. This cavity is filled with a mixture of molten fuel and fission gas, and expands continuously, both radially and axially, as the fuel melts. This can occur in both metal and oxide fuel pins, although fuel relocation and melt progression are different due to the thermo-physical properties of these fuels. In metal fuel pins, with a higher thermal conductivity, the axial temperature profile peaks near the top of the active core, and the molten cavity tends to develop near the top of the pin. Thus, in metal fuel cores, it is likely that the molten cavity will reach the top of the fuel column prior to the occurrence of cladding failure. This can also occur in oxide cores, where the molten fuel cavity tends to develop closer to the core mid-plane. If no blanket pellets are present, the pressurized molten fuel in the cavity can relocate rapidly to the lower pressure upper plenum, introducing a substantial amount of negative reactivity and causing an associated power decrease. If cladding failure occurs after the initiation of the in-pin molten fuel relocation, it is likely to happen at lower reactivity and power levels, an important safety advantage during the early stages of molten fuel relocation. If, on the other hand, the cladding failure occurs prior to the onset of in-pin fuel relocation, it will lead to a rapid cavity depressurization and thus prevent a later fuel ejection to the space above the fuel column.

After the occurrence of cladding failure the fuel reactivity feedback is the net result of the in-pin and ex-pin fuel relocation events. The early post-failure fuel relocation is dominated by the rapid acceleration of the in-pin molten fuel towards the failure location. If the failure location is near the core mid-plane, the in-pin molten fuel is relocated towards the higher reactivity region, and can lead to a temporary net positive fuel reactivity feedback. If the initial cladding failure is located further above the core mid-plane the initial in-pin fuel motion tend to move at least some of the fuel towards regions of lower reactivity and the net fuel reactivity feedback due to the early fuel relocation can become negative.

A significant safety advantage of the metal fuel cores is that the physical properties of the metal fuel, in particular the high thermal conductivity, lead to an initial cladding failure located above the core mid-plane for a wide range of postulated severe accident situations. The molten fuel ejected into the coolant channel is generally relocated towards the lower reactivity regions of the core by the pressure gradient and thus provides a negative reactivity feedback. Shortly after the cladding failure, once enough molten fuel has been ejected into the coolant channel and has been accelerated towards the core periphery, the negative reactivity feedback due to the coolant channel fuel dispersal begins to dominate, and the net fuel relocation reactivity feedback becomes strongly negative.

In contrast to the LOF accident, where fuel melting and relocation can begin within tens of seconds following the accident initiation if the accident initiators are rapid enough, the sequence of events in a degraded decay heat removal transient develops over a period extending from several hours to several days. If the auxiliary decay heat removal system functions as designed, fuel melting will not occur. However, if the reactor vessel decay heat removal system function is also substantially degraded, gradual melting of the core will eventually occur after boil-off of the sodium. Early power excursions are precluded in this case since the reactor is highly subcritical, and the possibility of energetic secondary re-criticality events caused by fuel relocation and compaction are the main concern.

There are two limiting scenarios that describe the molten core debris flow downward toward the lower plenum during a LOF event. If the structure below the core has large diameter coolant channels, freezing and plugging would not substantially obstruct the downward flow and the molten core material could flow rapidly from the core, reaching the lower core structures in a short time after cladding failure. If the lower core structure, however, has very small diameter channels with a substantial heat capacity then the core debris could freeze and plug the channels.

For the case with large diameter channels and if the coolant channels in the core have been voided, the melt can descend rapidly into the sodium filled areas below the core, driven by fission gas pressure and, in oxide fueled reactors, steel vapor pressure. The result of the melt-sodium contact is a function of the vigor of the melt contact with the sodium and on the temperature of the molten core materials at the time of contact. In the case of oxide fuel, which has a much higher melting point and hence melt temperature than the sodium boiling point, the melt-sodium contact is expected to lead to solidification and fragmentation with very small particles. In the case of metal fuel, which melts slightly above the boiling point of sodium, the melt-sodium contact may result in incomplete fragmentation or fragmentation with very small particles. Such fragmentation may improve the lateral spreading once the materials have moved into more open areas, but the melt will still have a high rate of decay-heat generation.

For the case with small diameter channels, where freezing and plugging is likely to obstruct the coolant channels, the melt will probably emerge from the assembly by melt-through of the duct wall. In that case the flow will continue within the spaces between the ducts, with the fuel melt displacing sodium as it flows downward and may eventually melt through another neighboring duct wall. Considering the difficult downward path for the melt and the low temperatures in the regions below the core, it will take a relatively long time for the fuel to reach the core support plate and the decay heat level will be considerably lower at that time. The melt will enter the sodium gradually, leading to complete fragmentation with larger particles, but the degree of lateral spreading may be small. The melt will contain considerably more steel than in the previous case.

Once the core materials have moved down and out of the bottom of the assemblies, a debris bed would form in the inlet plenum. Experiments performed at ANL with metal fuel and sodium showed the formation of high porosity debris beds, but the porosity of oxide fuel beds is considerably lower. The heat generated in the porous bed is removed by conduction, convection without boiling, and eventually coolant boiling if the first two heat transfer mechanisms are not sufficient to cool the debris.

Assuming that a high-pressure loading due to an energetic power excursion does not occur, the most likely vessel failure mechanism would be by melting the steel vessel wall due to direct contact with a high temperature molten fuel/steel mixture. This can result in liquefaction of the steel if the fuel temperature is high enough. The normal melting point of the steel is around 1700 K. In an oxide-fueled reactor the fuel melting temperature is around 3000 K, considerably higher that the steel melting temperature. In a metal fueled reactor the fuel melting temperature is around 1400 K, lower than the steel melting temperature, but chemical interactions between fuel and the steel components are likely to lower the effective melting temperature of the steel. The rapid eutectic penetration temperature is approximately 1400 K. If the core materials are cooled in the inlet plenum, either as a particulate bed or a molten pool, then direct contact between the core fuel and the reactor vessel would be precluded and core materials would be contained in the inlet plenum of the vessel.

The processes that would occur after a melt-through of the core material out of the reactor vessel include the chemical reaction of the sodium with concrete, the release of water vapor from the concrete, the generation of hydrogen as a result of the sodium-water or sodium-concrete reactions, and the interaction of molten fuel with concrete. The principal concern with respect to off-site radiological consequences is the potential for loss of integrity of the containment envelope due to possible buildup of internal pressure, temperature, or explosive gases. Studies have indicated that, in the absence of an engineered out-of-vessel core catcher, fuel in contact with concrete would cause concrete melting and limited penetration, due to the mixing of the molten fuel with concrete and resulting dilution and heat flux decrease.

Containment building temperature and pressure transients are determined primarily by the chemical reactions of the sodium expelled from the reactor system and by the decay heat of released fuel and its chemical interactions with containment materials. The release of the resultant noble gases and aerosols depends on the mixing with the containment atmosphere and the leakage paths present. The halogens and solids are subject to removal mechanisms within the containment, primarily dependent on aerosol mechanics. Aerosol agglomeration, settling, and plate-out can remove particulate radioactivity with a removal time constant of the order of hours. Removal of aerosols in the leak paths can also play a significant role in reducing the release of radioactivity from the containment building.

In parallel, the development and validation of the SAS4A oxide fuel code was continued by an active collaboration involving Japan, Germany, France, UK, and the EU, and the SAS4A code continues to be used actively in those countries for the study of the initiating phase of postulated SFR severe accidents. Specialized modules of the SAS4A code were developed to describe the material relocation during UTOP and ULOF events, enabling whole-core analyses of the initiating phase of these accidents. Modules of SAS4A were modified to allow the modeling of metal fuel pins, and a new module was developed to describe the pre-clad-failure in-pin fuel relocation. Research activities in Japan and France are currently focused on the development of the SIMMER-III and SIMMER-IV codes for the analysis of the transition phase for oxide-fueled reactors, with the goal of carrying out integrated initiation phase and transition phase analyses. It is worth noting that transition phase analyses do not appear to be necessary for metal fuel designs.

Just as for the HTGRs, if MELCOR was selected by the non-LWR technical community as the SFR severe accident tool, new MELCOR models would be needed to be able to analyze both metal and oxide fuel designs. Such models would not only be for fuel motion, materials interactions, etc., but would also include estimating coefficients of reactivity and reactor kinetics behavior. Finally, interfacing models would be needed for input into SIMMER-III or SIMMER-IV. Prior to developing these models, a careful review of those used in SAS4A would be carried out.

Experimental Needs and Requirements

A literature review will be carried out to identify relevant HTGR, SFR, and MSR experiments.

In particular for HTGRs, it is likely that additional fission product release, transport, and deposition occurs during accident conditions. Because the release of non-fission product aerosols from the core increases fission product aerosol deposition, this literature review should include experiments on aerosol releases of other core materials under accident conditions. Based on the results of the literature review, additional experiments may be recommended.

Many experiments were carried out in the 1970s and 1980s for SFRs. They include experiments pertaining to sodium fires, sodium-water reactions, fuel failure propagation, energetic loadings on containments, and other severe accident-related issues. The evaluation of severe accidents has played a prominent role in the safety analysis of SFRs. During the licensing evaluations of FFTF and Clinch River Breeder Reactor extensive experimental and analytical work was performed in the US to evaluate the phenomena, accident path, and consequences of postulated accident initiators that could lead to core disruptive accidents. During the Clinch River Breeder Reactor project in the 70s and 80s, much research was done on severe accidents for SFRs with oxide fuel pins. The research included in-pile and out-of-pile experiments to study severe accident phenomena, and extensive code development effort to evaluate postulated severe accident consequences. With re-focusing of the US SFR program on metal fuel, experimental and analytical activities shifted to metal fuel phenomenology. A series of metal fuel experiments was performed, and these experiments were analyzed with the SAS4A-M (metal fuel) code.

At this time, it is not known what experiments would be required to carry out a molten salt severe accident review.

References

Nuclear Energy Agency, “Experimental Facilities for Sodium Fast Reactor Safety Studies, Task Group on Advanced Reactor Experimental Facilities (TAREF)” NEA/CSNI/R (2010)12, Paris, France, 2011.

IMPLEMENTATION ACTION PLANS: SEVERE ACCIDENT PHENOMENA

Develop the Capability to Perform Severe Accident Analysis of Gas-Cooled Reactors [e.g., Confirmatory Codes and Analysis for Severe Accident Analysis]

Contributing Activity No. 2.21: Develop severe accident analysis code applicable to gas-cooled reactors.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Review existing PIRT(s) for gas-cooled reactors by general type; thermal and fast, and modify as appropriate. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X	X	RES
FY18			
FY19			
FY20			
FY21			
Confirm adequacy of MELCOR as the staff tool for confirmatory analysis. This will be done collaboratively with the non-LWR technical community. The resources shown represent NRCrepresent NRC portion of necessary resources to complete this work work			
FY17			
FY18	X	X	RES
FY19			
FY20			
FY21			
Identify potential shortcomings in MELCOR and formulate a plan for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY17			
FY18			
FY19	X	X	RES
FY20			
FY21			
Perform code and model development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.22: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data as made available. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.		0	
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Perform assessment against available data. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: SEVERE ACCIDENT PHENOMENA RESEARCH

Develop the Capability to Perform Severe Accident Analysis of Liquid Metal-Cooled Fast Reactors [e.g., Confirmatory Codes and Analysis for Severe Accident Analysis]

Contributing Activity No. 2.23: Develop severe accident analysis code applicable to liquid metal-cooled fast reactors

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for LMFRs by general type; pool and loop. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17	X	X	RES
FY18			
FY19			
FY20			
FY21			
Review the available severe accident codes and evaluate for further development as the staff tool for confirmatory analysis. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18	X	X	RES
FY19			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
FY20			
FY21			
Identify potential shortcomings in the severe accident code and formulate a plan for further development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19	X	X	RES
FY20			
FY21			
Perform code and model development. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			
FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.24: Identify experimental data needs and begin code assessment

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data as made available.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Perform assessment against available data.			

FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

IMPLEMENTATION ACTION PLANS: SEVERE ACCIDENT PHENOMENA RESEARCH

Develop the Capability to Perform Severe Accident Analysis of Molten Salt Reactors [e.g., Confirmatory Codes and Analysis for Severe Accident Analysis]

Contributing Activity No. 2.25: Develop severe accident analysis code applicable to molten salt reactors

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop or obtain PIRT(s) for molten salt reactors.			
FY17	X	X	RES
FY18			
FY19			
FY20			
FY21			
Review the available severe accident codes and evaluate for further development as the staff tool for confirmatory analysis.			
FY17			
FY18	X	X	RES
FY19			
FY20			
FY21			
Identify potential shortcomings in the severe accident code and formulate a plan for further development.			
FY17			
FY18			
FY19	X	X	RES
FY20			
FY21			
Perform code and model development			

FY17			
FY18			
FY19			
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.26: Identify experimental data needs and begin code assessment.

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
Identify experimental information necessary for code assessment. Obtain data as made available.			
FY17			
FY18			
FY19	X	X	RES
FY20	X	X	RES
FY21			
Perform assessment against available data.			
FY17			
FY18			
FY19	X		RES
FY20	X	X	RES
FY21	X	X	RES

4.2.3.5 Functional Area: Offsite Consequence Analysis

Overview

Societal consequences for LWRs are calculated by the NRC, industry, and international organizations using the MACCS (MELCOR Accident Consequence Code System) computer code suite. MACCS models atmospheric transport and dispersion (ATD) of airborne radioactive plume segments, offsite protective actions, exposure pathways, health effects, and societal consequences including land contamination, relocated population, and economic cost. MACCS currently considers 87 parent and daughter radionuclides. The impact on offsite consequences depends on the radionuclides and their physical characteristics in terms of how they transport through the atmosphere, deposit on the ground, and how they are biologically modeled in the human body for modeling early and latent cancer fatality risks. Dose conversion factors and other biokinetic factors such as uptake in foodstuffs account for these chemical forms. If the non-LWR technical community selects MACCS as the analytical tool for offsite consequences for any non-LWR design type, and if biologically important radionuclides are produced in increased quantities for non-LWRs compared with existing designs, they must be added to the MACCS library. If new chemical forms are important, revised dose and uptake factors must be made available. Other analyses using a severe accident code would give a final list of the type and quantities of radionuclides produced, but MACCS would determine their biological importance.

Independent confirmation of risk will be used by the non-LWR technical community to evaluate the safety of the design and identify any risk outliers. For instance, a technical justification of the proposed size of the emergency planning zones (EPZs) might be needed. The supporting calculations should be commensurate with the calculations used in choosing the current 10-mile plume exposure pathway EPZ and the current 50-mile ingestion exposure pathway EPZ for operating LWR plants. NUREG-0654 (Federal Emergency Management Agency-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued March 2002, refers to these calculations. This document also discusses choosing the size of the EPZs. NUREG-0396 (EPA 520/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants," discusses these calculations more fully.

Offsite Consequences Codes and Methods

MACCS has primarily been used for analysis of offsite consequences from environmental releases at conventional LWRs. Non-LWR technologies may have different characteristics that warrant further evaluation and may require enhanced modeling capabilities. Additional information would be needed to determine whether the capabilities would be design type specific. Nonetheless, development of MACCS for non-LWRs needs to consider the following considerations:

- Radionuclides: MACCS currently considers 87 parent and daughter radionuclides that are considered the most important to offsite risk from LWRs. Non-LWRs use many different materials and therefore may contain additional radionuclides of importance that might need to be added to MACCS. The chemical form of a radionuclide is also important as it relates to both ATD and biological significance. NRC would need to evaluate if different chemical forms of radionuclides would be released from non-LWRs which might necessitate different modeling of atmospheric deposition velocities as well as biological uptake in the human body.

- Environmental Release Pathways: MELCOR calculates airborne source term releases of radioactive materials and MACCS models the ATD and offsite consequences of such airborne releases using a Gaussian plume segment model. If non-LWRs have greater potential for aqueous releases, the staff would need to evaluate methods for modeling them and their consequences.

- ATD Modeling: MACCS uses a Gaussian plume segment model for ATD in which many of the underlying parameters are based on data from experiments conducted in rural terrain (e.g., prairie grass experiments used for surface roughness correlation). If non-LWRs will be sited in more urban environments than large LWRs, some of the correlations may need to be updated to be more appropriate.

- Emergency Response: With non-LWRs desiring smaller EPZs than large LWRs, MACCS models may need updates to consider close-in population in greater detail than is done for a traditional 10-mile plume exposure pathway EPZ.

- Chemical Hazards: MACCS models radiological releases to the environment. If non-LWRs themselves, or because of their potential collocation with industrial processing plants, create greater likelihood of chemical releases to the environment, additional codes and models may be needed to also consider non-radiological public health impacts.

Specific work in the area of offsite consequence modeling will be necessary:

- Perform an initial scoping study identifying and prioritizing potentially relevant modeling needs;
- Obtain data on the radionuclides that would be released to the environment from the different non-LWR types including their physical, chemical, and biological properties;
- Based on the initial scoping study and design information available to date, implement needed modeling enhancements to be able to analyze offsite consequences for non-LWRs; and
- Conduct code verification activities and document the model developments and user guidance in the MACCS NUREG User Guide and Theory Manual.

Experimental Needs and Requirements

Experimental needs would be determined once more information has been developed.

4.2.8.4 IMPLEMENTATION ACTION PLANS: OFFSITE CONSEQUENCE ANALYSIS

Develop the Capability to Perform Offsite Consequence Analysis for Gas-Cooled Reactors, Liquid Metal-Cooled Fast Reactors, and Molten Salt Reactors [e.g., Confirmatory Codes and Analysis for Severe Accident Analysis]

Contributing Activity No. 2.27: Perform an initial scoping study identifying and prioritizing potentially relevant modeling needs.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Perform an initial scoping study identifying and prioritizing potentially relevant modeling needs. Obtain data on the radionuclides that would be released to the environment from the different non-LWR types including their physical, chemical, and biological properties. This will be done collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.	Mid-term activity, resources to be included in mid-term IAP		

Contributing Activity No. 2.28: Based on the initial scoping study and design information available to date, implement needed modeling enhancements to be able to analyze offsite consequences for non-LWRs.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Implement modeling enhancements, conduct code verification, and document code developments and user guidance in the MACCS NUREG User Guide and Theory Manual. This IAP assumes MACCS will be selected as the analysis code. The work will be done	Mid-term activity, resources to be included in mid-term IAPhours		

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
collaboratively with the non-LWR technical community. The resources shown represent the NRC portion of necessary resources to complete this work.			

4.2.3.6 Functional Area: Materials and Reactor Component Integrity

Overview

The assessment of materials and structural integrity issues during the licensing process to provide reasonable assurance of safe operation of any non-LWR will be unique to the specific design. The assessment will require knowledge of the materials, the environmental conditions, and the loading specific to each non-LWR design. Gas-Cooled Reactors, Sodium Fast Reactors, Molten Salt Reactors and other possible non-LWR designs will have unique new materials, and familiar LWR materials exposed to unique environmental conditions (or stressors), when compared to existing LWRs. The one commonality is that all the assumed non-LWR technologies operate at higher temperatures than conventional LWRs.

During past efforts to prepare for the possible licensing of a HTGR, a research plan was developed (NRC, 2011) and was being implemented to address associated materials and structural integrity issues. For the purpose of this IAP, it is assumed that issues for any future HTGRs would not differ significantly from those articulated in this research plan and that the needed staff and contractor expertise and technical skills will be consistent with this plan.

The initial focus will be to identify the material and structural integrity issues that need to be addressed during licensing, determine the confirmatory research needed to validate the industry approach to address these issues, assess the capabilities and needs of existing structural integrity tools to evaluate these issues, identify gaps in staff and contractor expertise and technical skills, and develop an approach to efficiently address these gaps. This effort will be facilitated, in part, through various domestic and international collaborations.

There is some international experience in HTGRs and SFRs (see Section 4.2 Strategy Overview). As part of this activity, collaborative efforts will be developed with appropriate technical and regulatory experts in countries, including Japan, Korea, France, Britain, and Germany, through the International Atomic Energy Agency (IAEA), Nuclear Energy Agency (NEA) of Organisation for Economic Co-operation and Development (OECD), and other direct agreements.

However, there is much that is currently unknown about the materials and structural integrity challenges for the other non-LWR designs. To address these challenges, DOE/NE Advanced Reactors Technology Program has initiated significant research activities and the American Society of Mechanical Engineers (ASME) Codes and Standards Committee has established a special division (Division 5) within Section III of the ASME *Boiler and Pressure Vessel Code* (BPV Code) to address technical areas related to high-temperature reactors, including material performance and qualification issues. The NRC staff is engaged in these activities.

Therefore, engaging with the DOE/NE research activities, ASME, and other domestic codes and standards activities is an important near-term activity that will be used to identify materials and structural integrity issues, ensure that potential regulatory implications are addressed prior to developing or endorsing standards, and provide needed licensing expertise to both NRC staff and contractors. However, such activities are not described in this section, but are discussed in the section for Strategy 4 of this report.

In the near-term, activities for this functional area will focus on technical gap assessment, planning, and skill and expertise development as appropriate. Mid-term (5 – 10 years) and longer term (> 10 years) activities will be required to implement the research plan(s) developed for viable non-LWRs and to support the development of a risk-informed, performance-based regulatory framework for these non-LWRs. Such activities will not be delineated until the viable non-LWR design(s) that will be licensed are identified.

Gas-Cooled Reactors

The outlet temperature (similar to core exit temperature in a LWR) of an HTGR can vary from as low as 650°C to 950°C or above. The high outlet temperature presents a number of challenges in the design and safety performance, particularly those associated with fuel and materials. The ultimate choice of outlet temperature can significantly affect the types of materials that would need to be selected to achieve appropriate safety margins. . With regard to material loads/stresses, while the primary system would be above atmospheric pressure, the lesser density of the coolant, when compared to LWR designs, would be expected to effect the assessment of materials and structural integrity issues.

Accordingly, the scope of initial R&D activities described in the existing research plan (NRC, 2011), primarily in the materials area, is predicated upon the choice of reactor outlet temperatures near the lower end (around 750°C) of the outlet temperature range. Two basic designs of HTGR were considered in developing the research plan, the graphite-moderated pebble-bed reactor and the prismatic-core reactor. Based on these assumptions, the plan identified safety-significant issues and phenomena, needs and gaps related to evaluation models and tools, data needs, and data sources for both high temperature metals and graphite and the development of structural integrity tools.

The principal new near-term activity is to evaluate the accuracy and continued viability of the existing HTGR research plan. This evaluation will determine if the initial assumptions are still applicable, and assess the technical progress made in addressing the issues and gaps identified in this plan. Of particular importance will be to determine if current plans for HTGR designs that may need to be licensed by the NRC are higher temperature (> 750°C) or will use significantly different designs than initially assumed. If the existing assumptions remain relevant, only minor plan revision should be necessary to account for technical progress since research was halted. If the existing assumptions are not relevant, the plan will be modified to address any gaps created by the new assumptions.

In subsequent years, starting in FY18, some research will be initiated that focuses on using existing information to begin initial modifications to structural integrity codes such that they could be used for limited feasibility and scoping studies. Information about materials performance, component and structural design, operating conditions, and loading history will be needed to begin such work. Initial experimental work will also begin by first upgrading, as needed, the testing infrastructure and then conducting some limited research to demonstrate the feasibility of the infrastructure. If enough certainty exists about the design, it may be possible to initiate limited confirmatory testing to address high priority issues.

The other component of near-term work is to continue to support the development of a draft regulatory framework for materials-related issues (relevant SRP chapters, guidance, etc.) for non-LWRs, and more specifically, HTGRs. The applicability of the existing regulatory framework to non-LWRs will be assessed to identify aspects that require revision as well as areas that need to be added. Some work will be initiated in the near-term to proceed with modifications and enhancements that are generic and not specific to any particular design.

To implement this research plan and the continued development of the regulatory framework in both the medium and long-term, DOE and/or an applicant will need to select a particular design. At this point, the details of the planned research will be revisited and refined as necessary. The plan will continually be assessed and refined as necessary as other major design choices (e.g., outlet temperatures) and material selection become more clearly defined.

Sodium-Cooled Fast Reactors

SFRs generally operate in the 480–540℃ (900–1000℉) coolant outlet temperature range near atmospheric pressure. The sodium coolant has a density similar to that of water, which could make the assessment of material loads/stresses similar to LWRs. The sodium coolant would, however, react exothermically with air and water. Accordingly, SFRs are designed to reduce the probability and consequences of such reactions. SFRs generally have a secondary/intermediate sodium flow loop which is connected to the primary sodium loop via a heat exchanger. The secondary/intermediate loop separates the primary loop, containing radioactive sodium from the steam or water working fluid in the tertiary loop of the balance of plant system. SFRs also do not have a conventional emergency core cooling system. Instead, to prevent a loss of coolant that could result in exposure of the fuel in the core, SFRs are designed with a second vessel which is referred to as a “guard vessel.” The guard vessel surrounds the reactor vessel (and for loop plants the guard vessel also surrounds the primary system pumps and heat exchangers).

Near-term research activities for SFRs will focus on identifying materials issues that are of safety significance, assessing the needs for structural integrity analysis tools, developing staff and contractor expertise and infrastructure, developing a research plan, and assessing the regulatory framework. There is both U.S. and international experience with licensing and operating commercial SFRs and this experience will first be mined and compared with plans for new SFR designs to determine relevancy. It is also planned initially to query material and SFR experts (using a process such as expert elicitation or the phenomena identification and rank table (PIRT) technique) to identify important material and structural integrity issues, current knowledge of these issues, and knowledge gaps that will need to be addressed by an applicant.

The information mined from operating and licensing experience and expert advice will be used as the basis to develop a research plan to identify the confirmatory work needed to assess material performance and also to update existing structural integrity tools for assessing the acceptability of the planned design and operation of safety-significant components and structures. The development of this research plan should be completed within 1 to 2 years of initiating this work.

In subsequent years, starting in FY18 as the research plan is being finalized, limited research will be initiated to begin modifications to structural integrity codes and upgrade the experimental infrastructure to perform subsequent confirmatory testing and develop data to support structural integrity code development. If enough certainty exists about the design, research could be started with the focus of preliminary verification and validation of the structural integrity codes. It may be also possible to initiate limited confirmatory testing to address high priority material performance issues.

Similar to the HTGR activities, the other component of near-term work is to continue to support the development of a draft regulatory framework for materials-related issues (relevant SRP chapters, guidance, etc.) for non-LWRs, and more specifically, SFRs. The applicability of the existing regulatory framework to non-LWRs will be assessed to identify aspects that require revision as well as areas that need to be added. Some work will also be initiated in the near-term to proceed with modifications and enhancements that are generic and not specific to any particular design.

To more fully implement the research plan developed for SFRs and to complete development of the regulatory framework in both the medium and long-term, DOE and/or an applicant will need to select a specific design. At this point, the details of the planned research will be revisited and refined as necessary. The research plan will also continually be assessed and refined as necessary as other major design choices and material selection become more clearly defined.

Molten Salt Reactors

Molten Salt Reactors (MSRs) are nuclear reactors that may use a fluid fuel in the form of very hot fluoride or chloride salt instead of the solid fuel used in most reactors. Such molten salts would be expected to have densities greater than water, which would be expected to effect the assessment of materials and structural integrity issues. The use of soluble fuel may also create unique environmental challenges for materials throughout the primary system regarding radiation exposure, as opposed to LWR designs where such concerns are generally limited to the reactor vessel and reactor internals. These types are reactors are designed to operate at atmospheric pressure and have no chemical reactivity with air or water. Similar to HTGRs, a relatively wide operating temperature is possible, with temperatures from $500^{o}C$ to $1000^{o}C$ reported depending on the specific MSR design.

As with the other reactor technologies, near-term research activities for MSRs will focus on identifying materials issues that are of safety significance, assessing the needs for structural integrity analysis tools, developing staff and contractor expertise and infrastructure, developing a research plan, and assessing the regulatory framework. While no MSR has been built, there has been research and design related to the MSR concept both in the U.S. and internationally. This information will be collected and evaluated initially to improve staff knowledge and expertise, identify the current progress and state-of-the-art with respect to material and structural integrity issues, and identify existing issues and the next steps required to address these issues.

The next step will be to query material and MSR experts (using a process such as expert elicitation or the phenomena identification and rank table (PIRT) technique) to identify important material and structural integrity issues, current knowledge of these issues, and knowledge gaps that will need to be addressed by an applicant. Because of the wide range of potential operating temperatures, the issues will need to be addressed for separate temperature regions where distinct materials and the degradation mechanisms are applicable. It would be extremely valuable to narrow the design temperature range expected for a potential applicant at this stage if that information is known.

The information mined from past research and expert advice will be used as the basis to develop a research plan to identify the confirmatory work needed to assess material performance and also develop or update existing structural integrity tools for assessing the acceptability of the planned design and operation of safety-significant components and structures. The development of this research plan should be completed with 1 to 2 years of initiating this work.

Starting in FY18, as the research plan is being finalized, limited research will be initiated to begin modifications to structural integrity codes and upgrade the experimental infrastructure to perform subsequent confirmatory testing and develop data to support structural integrity code development. If enough certainty exists about the design, research could be started with the focus of preliminary verification and validation of the structural integrity codes. It may be also possible to initiate limited confirmatory testing to address high priority material performance issues if enough design specificity exists at this time.

Similar to the HTGR activities, the other component of near-term work is to continue to support the development of a draft regulatory framework for materials-related issues (relevant SRP chapters, guidance, etc.) for non-LWRs, and more specifically, MSRs. The applicability of the existing regulatory framework to non-LWRs will be assessed to identify aspects that require revision as well as areas that need to be added. Some work will also be initiated in the near-term to proceed with modifications and enhancements that are generic and not specific to any particular non-LWR or MSR design.

To more fully implement the research plan developed for MSRs and to complete development of the regulatory framework in both the medium and long-term, DOE and/or an applicant will need to select a particular design. At this point, the details of the planned research will be revisited and refined as necessary. The research plan will also continually be assessed and refined as necessary as other major design choices and material selection become more clearly defined.

References

NRC (2011), "High Temperature Gas-Cooled Reactor (HTGR) NRC Research Plan," ADAMS ML103560222, January 2011.

IMPLEMENTATION PLAN FOR MATERIALS RESEARCH

Develop the regulatory framework to address potential materials issues related to specific non-LWR designs (high-temperature gas reactors (HTGR), sodium-cooled fast reactors (SFR), molten salt reactors (MSR) etc.), with a focus on issues that will provide the largest benefit per effort spent.

Contributing Activity No. 2.29: Assess the performance needs and issues for structural materials to be used in non-LWRs, such as HTGR, SFR, MSR. The assessment will include the state-of-the-knowledge, ongoing domestic and international research, applicable international OpE, codes and standards activities, gaps in knowledge, data, and assessment tools.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify major materials issues related to non-LWRs. Adopt PIRT –like process. (Literature review, collaboration and information sharing with DOE/NE, EPRI, international regualtory partners, contract support). Deliverable : Assessment Report			
FY17	X	X	RES/NRO
FY18	X	X	RES/NRO
FY19			
FY20			
FY21			
Identify the need for additional materials test data for developing confirmatory assessment tools (e.g. probabilistic fracture mechanics flaw evaluation, pressure-temperature limits, irradiation/environmental damage of materials)			
FY17	X		RES
FY18	X		RES/NRO
FY19			
FY20			
FY21			
FY21			

Note: The contributing activity to engage in domestic codes and standards activities related to reactor materials requirements and structural designs, such as facilitating the

development of ASME BPV Code, Section III, Division 5, is addressed by the IAP for Strategy 4.

Assumptions:

- Update operational (test and commercial reactors etc.) experience and international experience database for non-LWR materials issues.
- Develop collaborative initiative with DOE materials experts.
- Develop outreach initiatives with international entities (Multinational Design Evaluation Program (MDEP), Japan, France, Korea, UK, Germany) for information sharing and possible joint research activities on materials issues related to non-LWRs.
- Engage with ANS on General Design Request initiatives affecting advanced reactor materials.
- Identify and gather information on non-metallic materials for potential use (structural or fuel-cladding applications) in non-LWR technologies, such as carbon-carbon composites, ceramic-ceramic (cercer) and ceramic-metal (cermet) composites, and new metal alloys.

Contributing Activity No. 2.30: Conduct research activities to develop technical bases to resolve major materials related issues. Collaborate with domestic (DOE, EPRI, vendors) and international regulatory partners [based on the recommendations from the assessment report from contributing Activity No. 2.29].

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
• Research activities supporting knowledge and data enhancement of environmental effects on reactor components (internals, piping, welds, coatings) at high temperatures. (corrosion, radiation effects) [Note: We expect to leverage resources with DOE and EPRI on research activities.]			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES
Research activities supporting the development/enhancement of confirmatory assessment computation tools (high temperature flaw evaluation, damage assessment) [Note: We			

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
expect to leverage resources with DOE and EPRI on research activities.]			
FY17			
FY18	X	X	RES
FY19	X	X	RES
FY20	X	X	RES
FY21	X	X	RES

Contributing Activity No. 2.31: Support the development of a draft regulatory framework for materials-related issues (relevant SRP chapters, guidance, etc.) for non-light water reactors

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
• Assess existing regulatory guidance applicable to non-LWRs. Based on the assessment identify chapters/sections the require revision (either for all non-LWRs or for specific technology). NRO and RES to draft chapters/sections as needed.			
FY17	X		RES/NRO
FY18	X		RES/NRO
FY19	X		RES/NRO
FY20	X		RES/ NRO
FY21	X		RES/ NRO

4.3 Strategy 3: Develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes

Strategy Overview

This strategy supports the NRC's strategic objective of optimizing non-LWR regulatory readiness. As shown in the NRC's vision and strategy for improving the agency's readiness to regulate non-light water reactor (non-LWR) technologies, the strategic objective for optimizing regulatory readiness is:

> Regulatory review processes are optimized when the resources of the NRC and potential applicants are efficiently and effectively used in a way that meets NRC requirements in a manner commensurate with the risks posed by the technology, that maximizes regulatory certainty, and that considers the business needs of potential non-LWR applicants. Additional options for long-range changes for non-LWR regulatory reviews and oversight that would require rulemaking will also be considered. Regulatory readiness includes the clear identification of NRC requirements and the effective and timely communication of those requirements to potential applicants in a manner that can be understood by stakeholders with a range of regulatory maturity.

The near-term strategy to achieve this objective is defined as follows:

> Develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes. This flexibility will accommodate potential applicants having a range of financial, technical, and regulatory maturity, and a range of application readiness.

The near-term activities described below can be used to support longer-term efforts to develop, as needed, a new non-LWR regulatory framework that is risk-informed, performance-based, and that features staff review efforts commensurate with the demonstrated safety performance of non-LWR technologies.

The IAP for addressing regulatory readiness consists of several contributing activities in areas such developing decision criteria, selecting and categorizing licensing basis events, and improving regulatory processes to support various stages of reactor design activities. Various non-LWR technologies and specific designs based on similar technologies are at different points in the development process. A representation of design processes from the Department of Energy (DOE) Order 413.3B, "Program and Project Management for the Acquisition of Capital Assets," is shown in Figure 1[1]. This figure provides a useful distinction between different phases of project development, critical decisions, and the associated interactions that might be expected between the NRC staff and designers.

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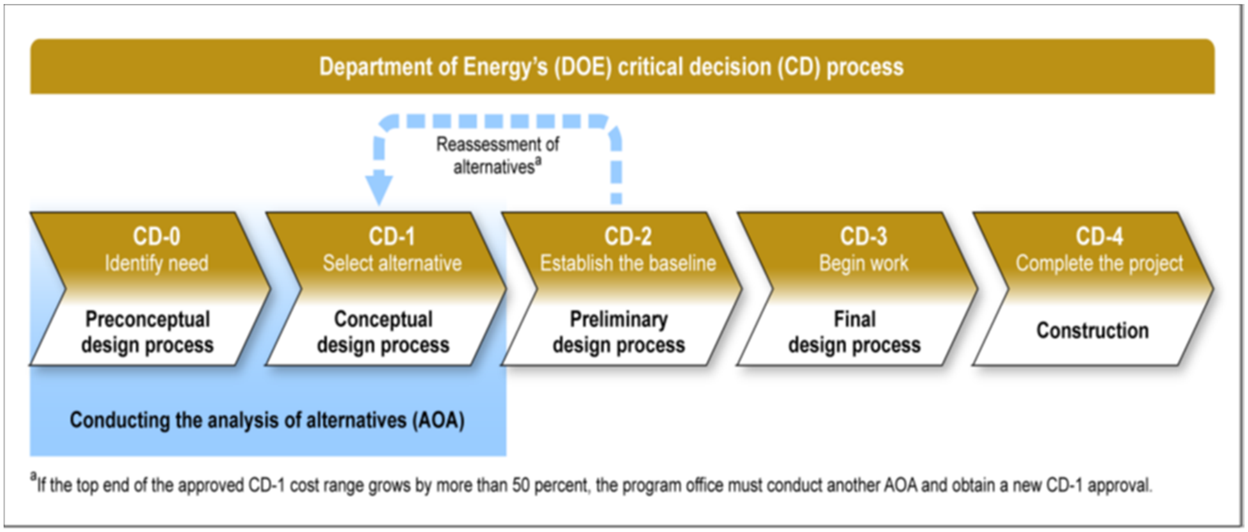
[1] From U.S. Government Accountability Office (GAO) Report GAO-15-37, "Analysis of Alternatives Could be Improved by Incorporating Best Practices," December 2014



Source: GAO analysis of DOE's Order 413.3B. | GAO-15-37

Figure 1: DOE Critical Decision Process

Current interactions between designers and the NRC range from activities in the preconceptual design process to designs in or nearly in the final design process. In addition, plans for the overall deployment of non-LWR designs might include multiple projects involving critical decisions for related research and test reactors, first-of-a-kind (FOAK) large scale plants, and subsequent commercial plants. The NRC's processes and practices need to be flexible enough to support interactions related to this wide variation in design development, recognizing that in some cases the NRC staff may be providing feedback and developing regulatory positions[2] in parallel with designers assessing various alternatives during the conceptual design process. The timing and scope associated with these regulatory interactions are intended to align with other related plans developed by external stakeholders working on non-LWR technologies. These related plans include plant design, research and development, finance, public policy, and fuel cycle.

The output from the IAPs will necessarily reflect the level of maturity of the various technologies and designs and is dependent on the designers, standards development organizations (SDOs), or other parties supporting the activities. The following sections discuss each of the contributing activities summarized in the attached IAP. Several of the contributing activities (e.g., decision criteria, licensing bases, and gap analyses) collectively establish a regulatory framework for a specific non-LWR technology and will be closely coordinated.

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2 In this context, "regulatory positions" may range from preliminary discussions with designers without the creation of documentation to be cited in future applications to Commission decisions (e.g., staff requirements memorandum or policy statement) or other published regulatory position (e.g., interim staff guidance, regulatory guide, or safety evaluation). Communications between the NRC staff and requester need to clearly define expectations for the interactions and the appropriate regulatory vehicles should be used to achieve the desired outcome (see Contributing Activity No. 3.4, Regulatory Roadmap).

Implementation Action Plans – Strategy No. 3

Contributing Activity No. 3.1 - Develop Decisionmaking Criteria

The NRC has developed detailed acceptance criteria supporting various regulatory decisions for LWR designs. Early reviews by the AEC and subsequent preliminary design reviews by the NRC also included assessing potential acceptance criteria for safety, security, and environmental reviews for several non-LWR designs, some of which were tested or commercially operated. In addition, some non-LWR designs have been completed and operated outside of the U.S. and information is available from other regulatory bodies, including the International Atomic Energy Agency (IAEA). An important supporting task for this activity consists of searching NRC records and other potential sources of information on regulatory approaches and acceptance criteria used for non-LWR technologies.

The acceptance criteria related to NRC's current regulatory requirements can be linked to physical phenomena related to water (e.g., departure from nucleate boiling), zirconium (e.g., metal-water reaction), and other materials and characteristics of LWR plants. Similar physical properties may likewise help develop acceptance criteria for non-LWR designs. The staff therefore plans to review available research, qualification testing, and other limitations that could help establish acceptance criteria (technology neutral or technology specific) for fuel forms, coolant systems, and other structures, systems and components needed to address key safety functions (reactivity, cooling, and limiting release). It is the Commission's expectation that non-LWR designs address the NRC's Advanced Reactor Policy Statement by providing longer time constants and using passive safety systems. Another part of developing acceptance criteria for such designs is ensuring that the regulatory requirements and associated NRC reviews are commensurate with the risks posed by designs developed in accordance with the policy statement. The related task for this activity will include consideration of experience with LWR SMR reviews, including design specific review standards and other review guidance, revised frameworks (e.g., NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing"), and hybrid approaches (e.g., NUREG-2150, "A Proposed Risk Management Regulatory Framework").

The Advanced Reactor Policy Statement includes as a goal for future plants that security be better incorporated into the plant design. Non-LWR plants would also expected to address potential aircraft impacts, mitigating strategies for various beyond-design-basis internal and external events, and other issues as much as possible as part of the plant design. Staff interactions with plant designers and other stakeholders should include identifying possible relationships and dependencies between regulatory areas such as safety analyses, probabilistic risk assessment, emergency preparedness, security, and environmental assessments. An integrated approach is important to enhancing plant safety as well as improving the efficiency of plant design, construction, and operation. The evaluation of dependencies and tradeoffs between plant designs and other requirements is also necessary to address stated goals for future plants, such as revising the emergency preparedness (EP) programs. This relationship is reflected in SECY-15-0077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies," dated May 29, 2015, which states:

> The staff anticipates that the technical basis for the EP framework would be developed as part of the rulemaking process and would include quantitative

guidelines and criteria for accident selection and evaluation, specific to SMRs and other new technologies. These guidelines and criteria would then be used to derive a dose-based, consequence-oriented rationale, similar to that described in SECY-11-0152, which would be used to inform the appropriate EPZ size for a specific design and its site. In addition, the staff would use historical and regulatory experiences gained over the past decades and insights gained from the results of using probabilistic risk assessment to inform the EP rulemaking. In addition to new regulations specifically addressing EP, the staff would expect to develop guidance for applicants.

Continued interactions with stakeholders, especially potential applicants through generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums, are essential to narrow and refine possible approaches and acceptance criteria for non-LWRs, technology groups, or specific designs. Lessons learned from large LWR designs and current activities for LWR SMRs will be used to help define appropriate balancing of design requirements and operational programs (i.e., use of performance-based approaches). The output from this activity is guidance documents describing licensing approaches and potential acceptance criteria for an integrated approach to the regulation and licensing of non-LWR designs.

Contributing Activity No. 3.1: Establish and document the criteria necessary to reach a safety, security, or environmental finding for non-LWR applicant submissions. The criteria and associated regulatory guidance are available to all internal and external stakeholders.

- Note that this activity is closely related to and will be pursued in parallel with Contributing Activities Nos. 2 and 3. The activity also includes, as a component in the overall development of decision-making criteria, the ongoing efforts to finalize the advanced reactor design criteria (ARDC).

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Conduct search of ADAMS, codes and standards (current and historical), IAEA, and other databases, and compile descriptions of licensing approaches and decision-making criteria for non-LWR designs (technology neutral or technology specific for subject non-LWR designs). Compilation may include possible approaches and acceptance criteria developed by the NRC staff, industry, or other stakeholders as part of this exercise.	2017		NRO, NRR, RES

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify physical phenomena, available research, qualification testing, or other limitations that could help establish acceptance criteria (technology neutral or technology specific) for fuel forms, coolant systems, and other structures, systems and components needed to address key safety functions (reactivity, cooling, and limiting release). Note that physical phenomena are likely to be technology and possibly design specific. The results of PIRTs completed under IAP Strategy 2 will be used to facilitate completion of this task.	2017-2018		NRO, NRR, RES
Evaluate and document possible means to develop licensing approaches and acceptance criteria to support regulatory reviews that are commensurate with the risks posed by non-LWR designs developed in accordance with the Advanced Reactor Policy Statement (e.g., longer time constants, passive safety systems) and that reflect key properties such as curie content, potential material forms, power level, and thermal capacities. Include consideration of experience with SMR design specific review standards, revised frameworks (e.g., NUREG-1860), and hybrid approaches (e.g., NUREG-2150).	2017-2019		NRO, NRR, RES, NSIR

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Identify possible relationships and dependencies between regulatory areas such as safety analyses, probabilistic risk assessment, emergency preparedness, security, and environmental assessments. Document findings and possible approaches to integrate and simplify regulatory requirements to address interfaces between various areas. See NUREG-0800 (SRP) Introduction, Part 2 for similar activity performed for light-water SMRs.	2017-2019		NRO, NRR, NSIR
Interact with potential applicants through generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums to narrow and refine possible approaches and acceptance criteria for non-LWRs, technology groups, or specific designs.	2018-2020		NRO, NRR
Activity Output: Prepare guidance document(s), possibly referencing consensus codes and standards or other references, describing licensing approaches and potential acceptance criteria for integrated approach to NRC requirements (e.g., relationships and trade-offs between designs, source terms, and emergency preparedness)	2020-2021		NRO, NRR, RES, NSIR

<u>Contributing Activity No. 3.2 - Develop Approaches to Licensing Bases</u>

A key aspect of the licensing of nuclear power plants is defining a logical approach for identifying and addressing various normal and off-normal events. These events are evaluated to ensure SSCs provide key safety functions to control reactivity, cool the core, and contain radioactive materials. Plant conditions and events have traditionally been classified as design-basis events (e.g., normal operation, anticipated operational occurrences, and postulated accidents) or beyond-design-basis events (e.g., station blackout and aircraft impact). As discussed in the previous activity, decision or acceptance criteria have been defined for the various categories of events. Event selection and classification approaches have been developed for non-LWR projects in the U.S. and in other countries. The first task planned by the staff is to conduct a search of NRC records, codes and standards, IAEA, and other databases and compile approaches for event selection; safety classification of SSCs; and other aspects of regulatory frameworks for non-LWR designs. A part of this activity is for the staff to document possible event types or categories (e.g., reactivity, loss of cooling, loss of inventory, chemical, natural external, security, etc.) that could challenge structures, systems, and components.

The NRC review of white papers related to the next generation nuclear plant (NGNP) project, the preapplication safety reviews of other non-LWR designs, and work with existing LWR plants have included discussions on the role of deterministic evaluations and probabilistic risk assessments within the process for selecting and analyzing potential events and accidents. Possible approaches are described within existing consensus codes and standards, IAEA publications, and NRC documents such as NUREG-1860 and NUREG-2150. The staff plans to interact with stakeholders, including potential applicants through generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums, to narrow and refine possible approaches to event selection and analyses. The output from this activity is guidance documents describing licensing approaches and potential event selection approaches for non-LWR technologies.

Contributing Activity No. 3.2: Determine and document appropriate non-LWR licensing bases and accident sets for highly prioritized non-LWR technologies.

- Note that this activity is closely related to and will be pursued in parallel with Contributing Activities Nos. 1 and 3.

Supporting Task Description	Job Hours Required	Contract Dollars	Participating Organizations
Conduct search of ADAMS, codes and standards, IAEA, and other databases and compile approaches for event selection, safety classification, and other aspects of regulatory frameworks for non-LWR designs (technology neutral or technology specific for subject non-LWR designs). Compilation may include possible approaches and acceptance criteria developed by the NRC staff, industry, or other stakeholders as part of this exercise.	2017		NRO, NRR, RES
Identify and document possible event types or categories (e.g., reactivity, loss of cooling, loss of inventory, chemical, natural external, security, etc.) that could challenge structures, systems, and components. Note that generic approach or technology specific approaches might be developed.	2017		NRO, NRR, RES, NSIR
Evaluate and document role of deterministic evaluations and probabilistic risk assessments in the identification and analysis of design-basis events and beyond-design-basis events (including potential severe accidents or equivalent). Note that generic approach or technology specific approaches might be developed. Include consideration of experience with design specific review standards, revised frameworks (e.g., NUREG-1860), and hybrid approaches (e.g., NUREG-2150).	2017-2019		NRO, NRR, RES, NSIR

Supporting Task Description	Job Hours Required	Contract Dollars	Participating Organizations
Interact with potential applicants through public meetings with generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums to narrow and refine possible approaches to event selection and analyses.	2018-2020		NRO, NRR, RES, NSIR
Activity Output: Prepare guidance document(s) describing regulatory requirements and event selection approaches to support integration of design and regulatory efforts (e.g., consideration of design basis events, loss of large areas due to fire or explosions, mitigation of beyond-design-basis events, etc.)	2020-2021		NRO, NRR, RES, NSIR

<u>Contributing Activity No. 3.3 - Identify Gaps in Regulatory Framework</u>

The differences between non-LWR designs and LWRs result in some predictable gaps between the existing NRC regulatory framework (developed primarily for LWRs) and the expected requirements to be established for non-LWRs. Evaluations of the applicability of existing requirements and the development of new requirements from the activities previously discussed have been performed for preliminary non-LWR designs and more recently as part of the NGNP project and for small modular light water reactor designs the staff is preparing to review. The staff will search NRC records and other databases to identify previously performed gap analyses for non-LWRs and light water reactors to take advantage of work previously completed including work done by DOE, IAEA and other international activities. For example, 20 non-LWR designs have been reviewed by the NRC and its predecessor the AEC to varying levels of completion including the PRISM, SAFR, MHGTR, Fort St. Vrain, Peach Bottom 1, Fermi 1, Clinch River, etc. The staff will use historical documents from these past reviews, comments associated with advanced reactor non-LWR design criteria (ARDC), and insights from ongoing activities to develop and maintain a list of non-LWR technical and policy issues that will need to be addressed in the near-term interactions with potential applicants and in defining and completing mid-term strategies (5-10 years). Note that the gap analyses are also closely related to the IAP for identifying and resolving policy issues (Strategy 5). The output from this activity will include guidance documents describing regulatory requirements, possible exemptions or exceptions, possible new requirements, and exceptions from or revisions to guidance documents (e.g., regulatory guides). Continued interactions with potential applicants through generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums are essential to identify and prioritize the evaluation of potential regulatory gaps (either exceptions from current requirements or defining needed new requirements).

Contributing Activity No. 3.3: Identify, document and develop plan to resolve current regulatory framework gaps for non-LWRs.

- Note that this activity is closely related to and will be pursued in parallel with Contributing Activity Nos. 1 and 2

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Conduct search of ADAMS and other databases (e.g., NGNP document library) to identify previous gap analyses for non-LWRs and document technical and policy issues. Compilation may include possible gaps identified by the NRC staff, industry, or other stakeholders as part of this exercise.	2017		NRO, NRR

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Use historical documents and comments associated with non-LWR general design criteria to develop and maintain a list of non-LWR technical and policy issues that will need to be addressed in the mid-term strategies (5-10 years) – **also see policy related IAP**	2018-2020		NRO
Interact with potential applicants through generic groups (e.g., NEI), technology groups, standards development organizations, and design specific forums to define and prioritize possible regulatory gaps for non-LWRs.	Captured in activities Nos. 1 & 2 (2018-2020)		
Activity Output: Prepare action plans describing regulatory requirements, possible exemptions or exceptions, and possible new requirements (from Contributing Activity Nos. 1 and 2) for subject non-LWR technologies. Note that results from this activity support potential longer term efforts to develop a new non-LWR regulatory framework (see Contributing Activity No. 3.7). Resolution of some gaps may come from other activities (i.e., decision making criteria) and resolution of some gaps may require developing separate action plan (similar to policy IAP).	2020-2021		NRO, NRR, RES

Contributing Activity No. 3.4 - Develop Regulatory Review Roadmap

The NRC staff should work with generic groups (e.g., NEI), technology groups, and individual designers to develop regulatory review “roadmaps” that reflect the design development lifecycle and appropriate points and methods of interaction with the NRC. A generic roadmap will be developed to standardize terminology and expectations. Technology or design specific licensing plans can then be developed in cooperation with groups or individual designers to align the regulatory review plan with other plans, including research and development. The staff plans to evaluate and document current descriptions from DOE, NGNP, IAEA, and other sources of relevant information to develop a representative lifecycle, such as represented in Figure 1. A key aspect of aligning the design, research, and regulatory processes will be including characterization of design or technology status (e.g., technology readiness level (TRL), phenomena identification and ranking table (PIRT) development for various technical areas). Examples of these relationships from DOE Guide 413.3-4A, “Technology Readiness Assessment Guide,” are shown in Figures 2 and 3.

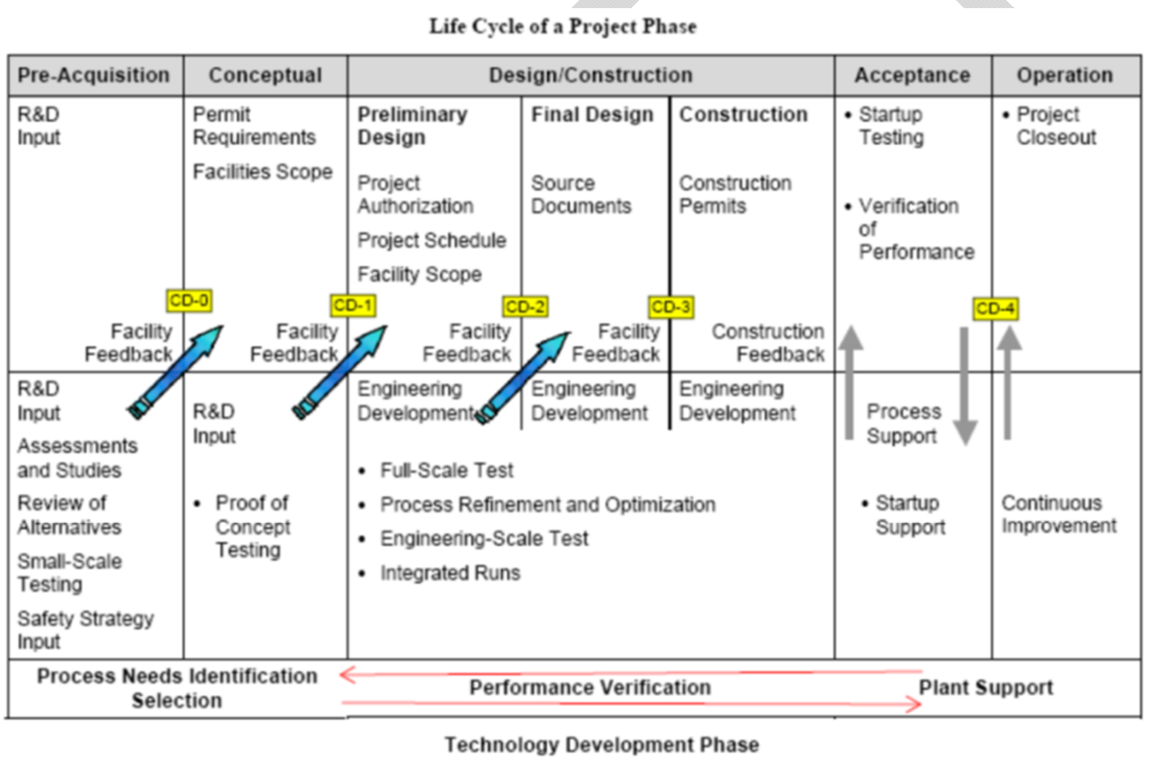

Figure 2: Technology Development Integration with Project Management

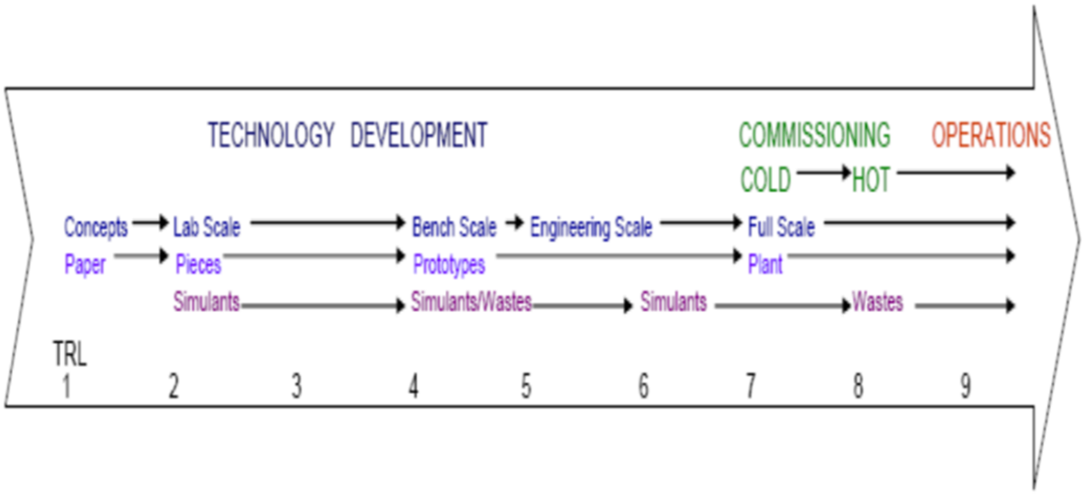

Figure 3: Schematic of DOE Office of Environmental Management
Technology Readiness Levels (TRLs)

The staff will document various available regulatory applications (e.g., CP/OL, SDA, DC, COL) and pre-application interactions (e.g., meetings, topical reports, white papers, conceptual design reviews) and describe how various tools can be used in various stages of the design process. A general summary of the processes is shown in Figure 4.

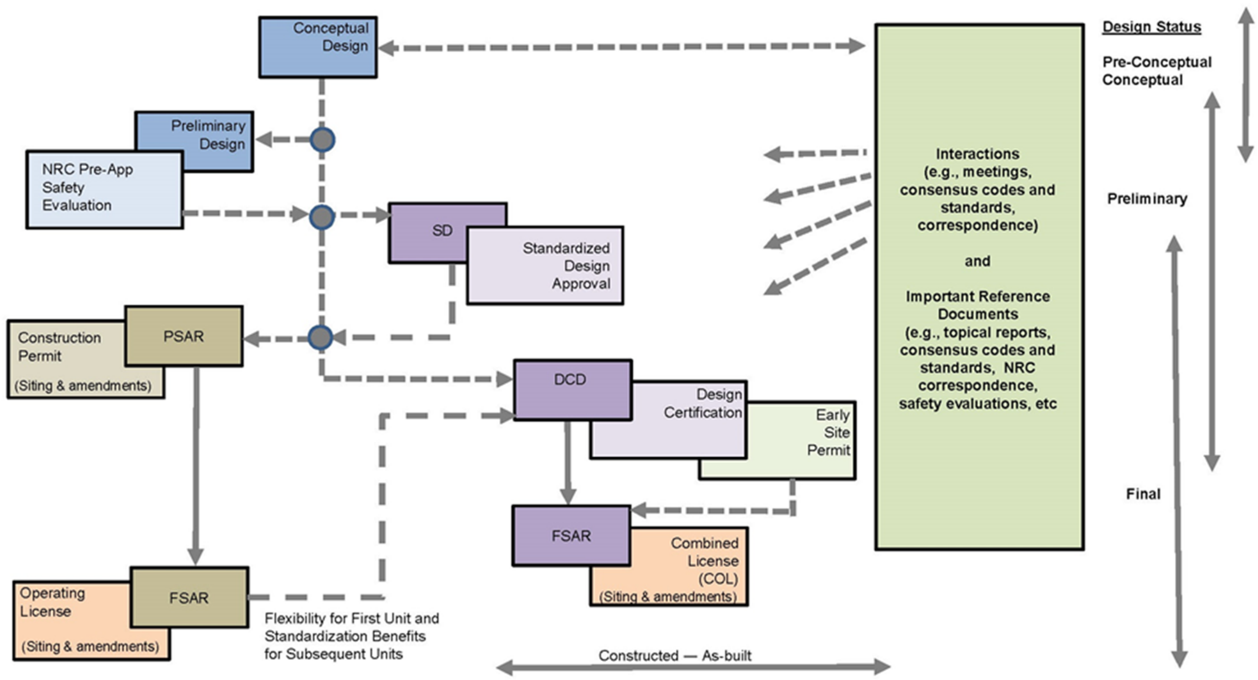

Figure 4: NRC Licensing-related processes

Previous preapplication interactions highlighted the importance of regulatory feedback in areas such as fundamental safety approaches, research, materials and fuel qualification. The staff included in the standard review plans for LWRs a related discussion of preapplication activities for light-water SMRs. The output from this activity will be a description of a flexible, non-LWR regulatory review processes, including the use of conceptual design assessments and standard design approvals, to define possible staged-review processes for designs or parts of designs at various levels of completion or confidence (e.g., across spectrum of TRLs). The alignment of regulatory interactions with other aspects of developing non-LWR designs requires a licensing plan (Contributing Activity No. 3.6) that reflects the results from assessments such as PIRTs or TRL evaluations (plant and/or SSC level); the need for and status of research and testing; and the prioritization of desired feedback from the NRC. The NRC staff and requester will need to agree on the appropriate levels of review and possible forms of feedback (e.g., verbal, correspondence, safety evaluation, etc.) considering available resources (NRC and requester), schedule, and importance. Aspects of the overall project plan dealing with the business model and some public policy issues may influence the priorities and schedules proposed by a designer but are not directly related to the NRC's regulatory review and licensing processes. NRC's ability to support the non-LWR program will be determined based on broader agency budgets and priorities. The roadmap will support the development of the technology- or design-specific licensing plans as described under Contributing Activity No. 3.6 in terms of defining interactions and processes and relationships between various stages of design, research and development, and licensing.

Contributing Activity No. 3.4: Develop and document a regulatory review "roadmap" that reflects the design development lifecycle and appropriate points of interaction with the NRC, and references appropriate guidance to staff reviewers and applicants.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Evaluate and document current descriptions from DOE, NGNP, IAEA, and other sources of the design development lifecycle, including characterization of design or technology status (e.g., technology readiness level (TRL), problem identification and ranking table (PIRT)). Identify from references or develop possible relationships between design development and other project-related models (e.g., financial, research and development).	2017		NRO, NRR, RES
Describe various available regulatory applications (e.g., CP/OL, SDA, DC, COL) and pre-application interactions (e.g., meetings, topical reports, white papers, conceptual design reviews) and provide guidance for these interactions (i.e., clarify expectations regarding submittals and corresponding NRC deliverables).	2017		NRO
Interact with potential applicants through generic groups (e.g., NEI), technology groups, and design specific forums to align regulatory processes with TRLs, research and development plans, and other aspects of the design development lifecycle.	2017		NRO, NRR, RES

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Activity Output: Guidance for a flexible non-LWR regulatory review processes, within the bounds of existing regulations, including the use of conceptual design reviews and standard design approvals, to define possible staged-review processes for designs or parts of designs at various levels of completion or confidence (e.g., across spectrum of TRLs). See SRP Introduction, Part 2 for related discussion of preapplication activities for light-water SMRs.	2017- 2018		NRO, NRR, RES

<u>Contributing Activity No. 3.5 - Develop Prototype, Research, and Test Reactor Guidance</u>

The possible commercial deployment of non-LWR technologies may be preceded by research or test reactors. In addition, even if deployment of a first-of-a-kind (FOAK) reactor for a particular design is a full scale commercial power reactor, it can be licensed as a prototype under 10 CFR 50.43(e)(2). Testing of the prototype reactor can be used to confirm integrated plant performance, analytical models, and any innovative means to accomplish key safety functions. The results of testing of the prototype reactor would be used to complement other research and testing and could be used within the licensing process to address uncertainties over the range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. Under paragraph 50.43(e)(2), the NRC may impose additional requirements on a prototype plant in order to protect the public and plant staff. These requirements would compensate for, among other things, technical uncertainties that exist before the testing program is complete and acceptable operation has been demonstrated. Examples of potential preventive and mitigative compensatory measures for a prototype plant include remote siting, supplemental robust systems, supplemental emergency preparedness measures, an incrementally staged startup process, limits on operating parameters imposed by technical specifications or license conditions, or a limited duration of the license. The possible licensing of a FOAK reactor as a prototype would be identified in the design-specific licensing plan described in Contributing Activity 3.6.

The staff plans to continue interactions with DOE as well as private sector groups and companies to help determine the expected role of prototype reactors, research and test reactors, and other potential vehicles to support the development of non-LWRs. The staff would, in turn, work with potential applicants, including DOE, to develop licensing plans for a materials test reactor and/or technology demonstration research/test reactor (RTR)[3]. DOE has indicated in its "Vision and Strategy for Development and Deployment of Advanced Reactors" that the role of the RTR(s) will be integral to the licensing and R&D plans for the associated technology or specific non-LWR design. If the NRC is to license the RTR, a regulatory review plan for the RTR should be developed given the licensing of an RTR is a major activity in and of itself. The staff will incorporate into the guidance insights from the recent issuance of a construction permit for a medical isotope facility. Therefore, it is important that the staff maintains an understanding of potential plans and identifies at the earliest possible time that a materials test RTR or technology demonstration RTR will require licensing by the NRC.

The output from this activity will be high-level guidance on the use of prototype reactors, research and test reactors, and other potential vehicles to address non-LWRs and the use of simplified, inherent, passive, or other innovative means to accomplish key safety functions.

Contributing Activity No. 3.5: Prepare and document updated guidance for prototype testing, research and test reactors.

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3 For this discussion, a materials test reactor (likely to be pursued by DOE) is primarily to provide an environment (fast neutron fluence, etc.) for the testing of fuels and materials for one or more non-LWR technology. A technology demonstration RTR is primarily to prove the performance of SSCs related to a specific design. The materials test reactor is likely to be a non-LWR design and may, therefore, also provide supporting information for a particular technology or design

- Note that draft guidance has been prepared related to the use of the prototype provisions defined in 10 CFR 50.43(e)

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Interact with potential applicants through generic groups (e.g., NEI), technology groups, and design specific forums to determine expected role of prototype reactor, research and test reactors, and the use of simplified, inherent, passive, or other innovative means to accomplish key safety functions.	2017-2021		NRO, NRR, RES
Interact with DOE, potential applicants, and technology/generic groups (e.g., NEI) to develop licensing plans for materials test reactor and/or technology demonstration research/test reactor(s). The role of the RTR(s) will be integral to the licensing and R&D plans for the associated technology or specific non-LWR design.	2017-2021		NRR, NRO RES
Activity Output: Complete and issue guidance on the use of prototype reactors, research and test reactors, and the use of simplified, inherent, passive, or other innovative means to accomplish key safety functions.	2019-2020		NRO, NRR, RES

Contributing Activity No. 3.6 - Develop Licensing Project Plans

This contributing activity relates to the development of technology- or design-specific licensing project plans[4]. Such licensing project plans would be inputs to applicants' overall development strategy addressing areas such as financing, research and development, fuel cycle, and public policy. The NRC's role and related regulatory review plan is focused on ensuring that potential design, construction, and operation of non-LWR technologies provide for the safe, secure and environmentally responsible use of radioactive materials. The development of technology- or design-specific plans under this activity would likely begin after progress has been made on other Contributing Activities, including the regulatory review roadmap (Activity 4). Technology groups or individual designers will inform the NRC when they are ready to more formally engage in preapplication interactions and develop a licensing plan. The integration of the licensing plan and other design development plans (e.g., research, finance) should enable the technology group or individual designer to better define priorities (e.g., what regulatory issue might drive costs that are key to project continuation or termination), schedules, and supportable costs for the regulatory interactions. The licensing plan is also an opportunity for the staff to develop regulatory approaches commensurate with the risks posed by the technology. The outputs from this activity will be a licensing project plan developed by the potential applicant and a corresponding regulatory review plan prepared by the NRC staff. The plans will identify the desired interactions; submittals and related NRC evaluations; dependencies on research and testing; costs and schedules; and other relevant information to allow applicants and the NRC staff to support the review. The roadmap prepared under Contributing Activity No. 3.4 will describe various types of interactions, submittals, and NRC deliverables. Periodic meetings and discussions between the staff and potential applicant provide opportunities to ensure the scope, schedule, and costs of activities remain consistent with the plans or to adjust the plans as appropriate.

Contributing Activity No. 3.6: Engage reactor designers and other stakeholders regarding technology- and design-specific licensing project plans and develop regulatory approaches commensurate with the risks posed by the technology.

- Note that interactions with reactor designers is an integral part of other contributing activities. This contributing activity relates to the development of technology- or design-specific licensing project plans. Such licensing project plans would be integral to an overall development strategy addressing areas such as financing, research and development, fuel cycle, and public policy. This activity would likely begin after progress has been made on other Contributing Activities and technology groups or individual designers are ready to more formally engage in preapplication interactions.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Support interactions with potential applicants to develop	2018-2021		NRO, NRR, RES

44 The licensing project plans discussed here are similar to and generally serve the same purpose as those described in the report, "Enabling Nuclear Innovation, Strategies for Advanced Reactor Licensing," Nuclear Innovation Alliance, April 2016. The NRC staff would develop an associated regulatory review plan to define the NRC's activities, outputs, and schedules for interactions included in a designer's licensing project plan.

licensing plans, including identification of reactor-design technical issues.			
Support interactions with potential applicants to identify siting, fuel cycle, or other technical issues.	2020-2021		NRO, NRR, RES

<u>Contributing Activity No. 3.7 - Support Longer-Term Regulatory Framework Development</u>

The near-term activities described above can be used to support longer-term efforts to develop, as needed, a new non-LWR regulatory framework that is risk-informed, performance-based, and that features staff review efforts commensurate with the demonstrated safety performance of non-LWR technologies. As such, the staff will document possible approaches and improvements for a possible new non-LWR regulatory framework during the development of guidance related to use of current regulatory framework.   The tracking of issues and potential improvements will be initially supported by available tools (e.g., SharePoint) and will be documented and incorporated into mid- and longer-term activities if or when they are pursued.

Contributing Activity No. 3.7: Support longer-term efforts to develop, as needed, a new non-LWR regulatory framework that is risk-informed, performance-based, and that features staff review efforts commensurate with the demonstrated safety performance of the non-LWR NPP design being considered.

Supporting Task Description	Job hours Required	Contract Dollars, $K	Participating Organizations
Activity Output: Document possible approaches and improvements for a possible new non-LWR regulatory framework during the development of guidance related to use of current regulatory framework	2017-2021		NRO, NRR, RES

Strategy 3 Assumptions:

The activities are focused on reactor design issues. Licensing project plans (technology or design-specific) may include siting issues, fuel cycle, and other non-design issues. The near-term focus will primarily be on reactor design, fuel design and qualification, and development of revised regulatory framework(s).

Activities proposed in the IAPs for near-term strategies do not include resolution of specific technical issues or any associated potential rulemaking. Resolution of specific technical issues would more likely fall under design- or technology-related project plans. Resolution of policy issues and potential rulemaking will instead be included in specific policy-related actions plans and/or the IAPs for the mid- or long-term strategies.

The activities should focus on preparations for SFRs, HTGRs, and MSRs. Note that SFRs and HTGRs are further along in the staged process, having been subject to previous pre-application interactions.

4.4 Strategy 4: Facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials)

Strategy Overview

This strategy supports the NRC's strategic objectives of enhancing technical readiness and optimizing non-LWR regulatory readiness.

For many years, the commercial nuclear power industry in the United States has been heavily focused on the development and operation of light water reactor technologies. As such, the development of codes and standards that have been endorsed by the NRC as part of its regulatory framework has also focused on light water reactor technologies. A partial list of codes and standards traditionally used by the NRC can be found on the public website (http://www.nrc.gov/about-nrc/regulatory/standards-dev/consensus.html). NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," and NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan," also contain references to codes and standards that the NRC staff uses to conduct its safety, security, and environmental reviews. Due to recent interest in the deployment of non-LWR technologies, it has become necessary for the NRC to consider adapting its regulatory framework to continue to ensure that these new and innovative designs are constructed and operated to protect public health and safety and the environment. In line with current practice, it is expected that the use of codes and standards will be an integral part of the NRC's strategy to improve its readiness to regulate non-LWR technologies.

As stated in NRC Management Directive (MD) 6.5, "NRC Participation in the Development and Use of Consensus Standards," it is the policy of the NRC to (i) involve all interested stakeholders in its regulatory processes, (ii) participate in the development of consensus standards that support the NRC's mission, and (iii) use consensus standards developed by voluntary consensus bodies consistent with the provisions of the National Technology Transfer and Advancement Act of 1995 and OMB Circular A-119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities." In line with this policy, the NRC has an established process (see MD 6.5) for implementing codes and standards into its regulatory framework. This process can be described in three primary steps, which are discussed in brief detail below.

(1) Identifying and Prioritizingthe Need for New and Revised Technical Standards

NRC offices conduct an ongoing review to determine whether there is a need to update or incorporate new codes and standards to address specific technical issues, new technologies, or regulatory processes. The staff identify and evaluate additional codes and standards that also may be endorsed as alternatives to existing regulatory requirements or guidance. If the staff identifies ongoing or planned SDO initiatives to develop a new standard in a relevant technical area, the staff considers the time frame in which the standard is needed in order to determine whether to participate in development of the applicable codes and standards.

Discussions with the relevant SDO(s) and other stakeholders are also an important part of the NRC's decision to use a code or standard. In addition to understanding the SDO's interest in developing or revising the standard and the time frame for the project, it is important for the NRC to determine the most effective committee level(s) for staff participation in order to make the best use of resources.

Some SDOs with which the NRC staff routinely interacts include ASME, ANS, IEEE, ASTM, IEC, and NFPA.

(2) Participation in Codes and Standards Development

Similar to all activities associated with the NRC's Vision and Strategy for non-LWR readiness, staff participation in codes and standards development should be in accordance with the NRC Principles of Good Regulation—independence, openness, efficiency, clarity, and reliability. Staff who participate in codes and standards activities on behalf of the NRC do so as specifically authorized representatives and are expected to follow these guidelines (MD 6.5).

While it is the NRC's responsibility to work effectively with all stakeholders, to clearly communicate its requirements, and to provide regulatory information and feedback in a timely manner, it is also the NRC's' responsibility to remain objective in its decision making and to involve the public in its regulatory processes. Codes and standards committee meetings are generally open to the public and the NRC staff's participation in such meetings does not necessarily imply NRC agreement with, or endorsement of, decisions reached by such organizations. Codes and standards are not approved for use within the NRC regulatory framework until they have been formally endorsed.

(3) Endorsement of Codes and Standards

NRC endorses consensus standards through incorporation by reference in regulations and through reference in such documents as regulatory guides, NUREG reports, and the standard review plans. In addition to consensus standards, the NRC may endorse international safety standards as acceptable means for meeting its regulatory requirements.

The NRC maintains its independence during participation in SDOs by reserving the right to apply conditions on codes and standards used in its regulatory process to ensure that they will meet the NRC's requirements to protect the public health and safety and the environment. The need to impose conditions may, however, be reduced by attempts to resolve outstanding issues through meetings with SDOs and other stakeholders, and active participation during the codes and standards development process.

As shown in the following IAP, the staff intends to support achievement of the strategic objectives by applying the established process for implementing codes and standards into NRC's regulatory framework.

Implementation Action Plan – Strategy No. 4

Contributing Activity No. 4.1 - Work with stakeholders to determine the currently available codes and standards that are applicable to non-LWRs and their associated fuels and waste, and to identify the technical areas where gaps may exist.

The goal of this activity is to (1) determine whether the appropriate codes and standards are available to support the safe construction of non-LWRs, fabrication of fuels, and subsequent waste management; (2) identify the codes and standards that may be endorsed in the future to support the non-LWR regulatory framework; and (3) encourage stakeholders (DOE, vendors, the public, etc.) to actively participate in the development of the codes and standards applicable to their designs. The NRC will accomplish this goal through outreach with stakeholders including the public; SDOs; DOE; EPRI; International counterparts and organizations; and non-LWR vendors. Once the supporting tasks for this activity are complete, the NRC will be better able to allocate resources and effectively prioritize its future involvement in various codes and standards activities.

It is recognized that the commercial nuclear power industry is the primary driver for codes and standards development. Therefore, it is important for the NRC to encourage that the various external stakeholders (i.e. vendors, DOE, NEI, EPRI, the public, etc.) that have interest in the non-LWR industry are aware of and, when practical, actively involved, in the standards development process. For this reason, facilitating non-LWR stakeholder involvement (including the public) with SDOs is a necessary component of this action plan.

Contributing Activity No. 4.1 - Work with stakeholders to determine the currently available codes and standards that are applicable to non-LWRs and their associated fuels and waste, and to identify the technical areas currently supported by codes and standards (e.g., instrumentation and control, civil/structural, inservice inspection and testing, materials, equipment qualification, quality assurance) where gaps may exist.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Survey internal and external stakeholders to determine the following (this effort will be tied to the identification of safety, security, regulatory, and design issues of various reactor designs and PIRT analyses proposed in the technical/research IAPs): 1. Currently available (in draft or final form) codes and standards that are written for non-LWRs (and associated fuel cycles). 2. Currently available codes and standards that	FY17: X FY18: X		NRO, RES, NMSS

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
generically apply to non-LWRs as written. 3. Currently available codes and standards that are generic and may be applicable to non-LWRs after revision or changes to the endorsing document. 4. Currently available codes and standards that are considered applicable to non-LWRs, but are not adequate to support their deployment (e.g., new standard has to be developed, additional research is needed in specific technical areas) 5. Status of vendor involvement in the development of applicable codes and standards, particularly where the vendors have identified gaps for their design.			
Communicate with each SDO to determine if there are any ongoing or planned activities related to non-LWRs.	FY17: X FY18: X		NRO, RES, NMSS
Communicate with international counterparts to determine the international codes and standards that are used to support the deployment of non-LWRs in their respective countries. (Organizations/forums of interest include the Nuclear Energy Agency, especially CNRA and CSNI; GIF; MDEP, especially CSWG, DICWG, and VICWG; IAEA especially INPRO; NEI; WNA; EPRI; and other regulators with operating non-LWRs)	FY17: X FY18: X		NRO, RES, NMSS

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop as part of the Communication Strategy (IAP Strategy No. 6) methods to help facilitate and encourage non-LWR vendor involvement in the development of codes and standards applicable to their designs.	FY17: X FY18: X		NRO, RES, NMSS
Identify codes and standards for which the NRC should provide resources/participation	FY17: X		NRO, RES

Basis:

The level of resources needed to support this activity may change depending on the method of communication with the external stakeholders. For example, a Category 3 public meeting may be less resource intensive than several Category 1 public meetings.

When possible, the NRC should make use of established forums and meetings to communicate with stakeholders and gather information on codes and standards development. For example, RES has an established Standards Forum for communicating with all the SDOs of interest to the NRC.

In order to optimize resources, the NRC should make use of innovative telecommunication tools, when available, to communicate with international counterparts and SDOs located outside of the United States.

Contributing Activity No. 4.2 - Participate with the SDOs that are actively involved in developing codes and standards for non-LWRs

The goal of this activity is for staff to participate in the development of codes and standards that will support the safe and secure deployment of non-LWRs. Active involvement in the codes and standards development process will allow the staff to determine if, when, and by what means, it would be appropriate to incorporate the applicable codes and standards into the regulatory framework for non-LWRs. Participation in the codes and standards activities of interest will also facilitate a more efficient endorsement review by allowing the staff to identify, discuss, and resolve potential issues early in the standard development process. As previously stated, the NRC may apply conditions on the use of codes and standards that it uses in its regulatory process. The staff will also ensure that the NRC staff's acceptance of codes and standards for use (with conditions and limitations as appropriate) are consistent with staff positions.

Even in cases where the staff's concerns are ultimately not resolved by the SDO, the insights gained through participation in the standards development process may be useful in determining the appropriate conditions to apply.

Lastly, participation in codes and standards development may support activities outlined in other Implementation Action Plans (IAPs). For example:

- The staff may identify opportunities for training and development (Strategy No. 1)
- The staff may identify a need for computer codes/tools or future research (Strategy No. 2)
- The staff may identify issues that would require policy decisions (Strategy No. 5)
- SDO meetings provide an opportunity for the staff to communicate with various non-LWR stakeholders and to encourage active involvement in codes and standards development (Strategy No. 6)

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
NRC staff attend the applicable codes and standards meetings and actively participate in the development process.	FY17: X FY18: X FY19: X FY20: X FY21: X		NRO, RES, other offices as applicable
For each standard, determine the potential methods of endorsement (e.g., incorporation by reference in the regulations, endorsement in a RG, specific approval in the context of an application) and determine milestones for when the NRC should consider starting a formal review.	FY17: X FY18: X		NRO, RES, Other participating organizations will depend on the standard of interest

Basis:

The level of resources (i.e. job hours, FTE, contract dollars, and travel dollars) needed to support this activity is directly dependent on the codes and standards of interest and the level of effort needed for the staff to participate in a manner that would benefit the NRC. Therefore, it is expected that resource estimates will be made on a case-by-case basis as the applicable codes and standards are identified in Contributing Activity No. 4.1. This will allow the staff to prioritize activities and make the best use of available resources. It is expected that each of the applicable codes and standards will fall into one of the following categories:

- Currently available (in draft or final form) codes and standards that are written for non-LWRs.
- Currently available codes and standards that are generic and apply to non-LWRs as written.

- Currently available codes and standards that are generic and may be applicable to non-LWRs after revision.
- Currently available codes and standards that are considered applicable to non-LWRs, but are not adequate to support their deployment (i.e. new standard has to be developed, additional work is needed in specific technical areas)

To support this activity, the NRC should allocate resources for active participation in the development of only those codes and standards for which meaningful progress has already been made to develop rules for non-LWRs. In other cases, the NRC should allocate resources to only track the progress of the development of a code or standard.

Contributing Activity No. 4.3 - Review codes and standards for endorsement

The goal of this activity is to review the codes and standards for endorsement by the NRC. This activity is written as a standalone effort, but in some cases, it may be more efficient to perform the review as part of the endorsement process (i.e. Regulatory Guide development or rulemaking, which often are complementary). To make the best use of resources, the NRC will not consider a formal review of any code or standard that is incomplete, has significant unresolved technical issues, is in contrast to standing NRC positions/policies (e.g., is solely risk-based), or would provide no benefit to the non-LWR regulatory framework.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
For each standard, determine an estimate of the resources required to review the standard.	FY17: X FY18: X FY19: X		NRO, RES, other offices as applicable
For each standard, identify the technical staff and develop the schedule to perform the review.	FY17: X FY18: X		NRO, RES, other offices as applicable
Perform review Note: The RES estimates assume the review of standards for materials (metallic, ceramic, composite, and cementitious/concrete), digital I&C, cables, and seismic/structural for only RES. These estimates are separate from the Other Offices' Estimates. Actual total resources needed will depend on the scope/level of review, and the number of standards which is known from	NRO: X (over 5 yrs) RES/DE Estimates: FY17: X FY18: X FY19: X FY20: X FY21: X		NRO, RES, other offices as applicable

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
the previous contributing activities.			
RG Task (Mid-Term): Identify/issue/revise new RGs to endorse the needed codes and standards. This task will involve three steps which are described in the basis.	Mid-Term Activity – resource estimate will be included in Mid-Term IAP	0	RES/NRO, other offices as applicable

Basis:

The level of resources (i.e. job hours, FTE, and contract dollars) needed to perform the review is directly dependent on the Code and Standard to be reviewed. Therefore, it is expected that resource estimates to perform the review are made on a case-by-case basis.

To make the best use of resources, the NRC will not consider a formal review of any code or standard that is incomplete, has significant unresolved technical issues, is in contrast to standing NRC positions/policies (e.g., is solely risk-based), or would provide no benefit to the non-LWR regulatory framework.

<u>Regulatory Guidance Task:</u>

It is expected that, in addition to rulemaking, RGs will be used as a vehicle to endorse codes and standards as part of the non-LWR regulatory framework. The RG task describes the overall effort that may be required to identify, assess, and revise the applicable RGs to support the non-LWR framework related to codes and standards. The estimates assume that the development activities will occur at some level in the near-term, but it is likely that this will be a mid-to-long term task due to progress of the various SDOs and non-LWR vendors.

The first step would be to perform an initial assessment of the RGs to determine which RGs could be applicable to codes and standards related to non-LWRs. The NRC staff would make some assumptions about the non-LWR designs based on interactions with the vendors, and then categorize the applicability of the other Divisions.

The second step would be to categorize the level of effort required for the applicable RGs. For example, they could be applicable to non-LWRs: 1) with no modifications, 2) with some modifications, or 3) with significant modifications.

The third step would be to revise or issue new RGs. The projected resources for this activity will be addressed in the mid-term IAPs.

Overall Assumptions:

RES is the NRC's point of contact for codes and standards activities and has the resources available to identify and interact with the various SDOs. RES also maintains a list of all NRC staff participating in SDOs.

The terms "codes and standards" and "Standards Development Organizations" include both domestic and international entities. As such, the activities described in this IAP are applicable to both domestic and international codes and standards.

A primary assumption in developing this near-term IAP is that the vendors' concepts are mature enough to be at the point of considering the codes and standards needed for design, construction, and operation (i.e. the information is available at the time it is needed). Furthermore, even vendors with mature concepts may not have completed the assessment of the applicability of various codes and standards. Therefore, it is expected that for some codes and standards, Contributing Activity Nos. 2 and 3 may not begin until the mid-term (5-10 years).

Codes and standards development can be a multi-year process, so it is expected that for some codes and standards, Contributing Activity No. 4.2 may start in the near-term but continue into the mid-to-long term.

Some development of regulatory guidance, in anticipation of correlated rulemaking, is expected to begin in the near-term.

4.5 Strategy 5: Identify and resolve technology-inclusive policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs)

Strategy Overview

The identification and resolution of policy issues within the purview of the NRC contribute directly to regulatory predictability, effectiveness, and efficiency. Additionally, early identification and resolution of policy issues helps to achieve the agency's strategic objectives for non-LWRs: enhanced technical readiness, optimized regulatory readiness, and optimized communications.

Technology-inclusive issues; that is, those issues that apply widely to non-LWR designs independent of the specific technologies used, have the broadest applicability for the non-LWR regulatory framework.

Issues for non-LWRs can range from strictly technical issues that can be resolved in accordance with existing Policy, to technical issues that involve policy implications, to issues that are primarily matters of policy. The Commission maintains the ultimate say in determining when an issue needs to be decided by the Commission to establish a final agency position. The actionable steps outlined in this IAP will assist the staff and stakeholders in determining which past policies apply to non LWRs, whether there are new potential policy issues for non-LWRs to be examined, and will create/apply a more formal policy evaluation approach.

Implementation Action Plan – Strategy No. 5

Contributing Activity No. 5.1: Determine the applicability of previously identified policy issues to non-LWRs

The purpose of this contributing activity is to perform a comprehensive review of current and historical NRC policy-related recommendation and decision documents to form a current baseline for non-LWR policy matters. All work for this contributing activity is planned for FY2017.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Conduct reference search to identify documentation for previously identified policy issues and resolution activities (any reactor technology) and document the search bibliography	X		NRO, NMSS
Review existing SECY papers, Commission SRMs, Commission policy statements, and non-LWR review documents previously prepared by staff, such as preapplication safety evaluation	X		NRO, NMSS

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
reports, for discussion and/or disposition of non-LWR policy issues			
Review non-LWR knowledge management resources for identification of non-LWR policy issues	X		NRO, NMSS
Determine whether the policy issues identified above apply to non-LWR technologies generally	X		NRO, NRR, NSIR, NMSS, RES
Determine whether technology-inclusive policy issues that have been closed or resolved for LWRs, and that also apply to non-LWRs, should also be considered closed or resolved for non-LWRs	X		NRO, NRR, NSIR, NMSS, RES

Contributing Activity No. 5.2: Identify additional technology-inclusive policy issues for non-LWRs

The purpose of this contributing activity is to identify potential emergent, non-LWR technology-inclusive policy matters for further review and evaluation. Any identified issues will be added to the baseline for non-LWR policy matters. Additional potential non-LWR issues will be identified and reviewed annually.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Conduct one or more public meetings with NRC, DOE, DOE laboratory, and industry non-LWR experts to elicit, discuss, and document additional, previously unidentified, technology-inclusive non-LWR policy issues. Meeting to be held in FY2017	X	X	NRO, NRR, NSIR, NMSS, RES
Liaise with international stakeholders and non-LWR operators as possible to identify additional potential policy issues and resolutions based on international experience (FY2017)	X		NRO, OIP, NRR, NSIR, NMSS, RES

Staff review meeting and international feedback and finalize listing of additional issues for review (FY2017)	X		NRO, NRR, NSIR, NMSS, RES
Staff conduct annual emerging non-LWR policy reviews and incorporate results into agency planning and budgeting process (FY2018 – FY2021)	X		NRO, NRR, NSIR, NMSS, RES

Contributing Activity No. 5.3: Analyze and resolve technology-inclusive non-LWR policy issues identified in Contributing Activity Nos. 1 and 2

The purpose of this contributing activity is to perform the staff work necessary to develop recommendations for the Commission for all identified technology-inclusive policy issues, and to respond to Commission direction for the policy matters as directed. For the purposes of this IAP, six currently identified issues and six emergent issues are assumed in the near-term.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Perform an initial review of potential policy items (FY17)	X		NRO, NRR, NSIR, NMSS, RES
Prioritize policy issues for review and resolution (all FY17)	X		NRO, NRR, NSIR, NMSS, RES
Develop policy-specific action plans and then analyze and resolve policy issues, (FY17) – assume 3 issues @ 1,600 hrs ea	X		NRO, NRR, NSIR, NMSS, RES
Develop policy-specific action plans and then analyze and resolve policy issues, (FY18) – assume 3 issues @ 1,600 hrs ea	X		NRO, NRR, NSIR, NMSS, RES
Develop policy-specific action plans and then analyze and resolve policy issues, (FY19) – assume 3 issues @ 1,600 hrs ea	X		NRO, NRR, NSIR, NMSS, RES

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop policy-specific action plans and then analyze and resolve policy issues, (FY20) – assume 3 issues @ 1,600 hrs ea	X		NRO, NRR, NSIR, NMSS, RES
Develop policy-specific action plans and then analyze and resolve policy issues, (FY21) – if necessary, complete work on any remaining unresolved policy issues FY subtotal = 650 hrs	X		NRO, NRR, NSIR, NMSS, RES

Bases/Assumptions:

The contributing activities are inclusive of the entire fuel cycle. For example, the review of policy issues will include the review of policy issues for fuel cycle facilities and waste disposal.

While most activities proposed in the IAPs for near-term strategies do not include rulemaking, technology-inclusive policy resolution activities may include rulemaking during this period.

4.6 Strategy 6: Develop and implement a structured, integrated strategy to communicate with internal and external stakeholders having interests in non-LWR technologies

Strategy Overview

As shown in the NRC's non-LWR vision and strategy document, the strategic objective for optimizing communications is:

> The NRC will optimize its communication with non-LWR stakeholders by disseminating clear expectations and requirements for non-LWR regulatory reviews and oversight. These expectations and requirements will be expressed using multiple channels of communication appropriate to different stakeholder interests. NRC messaging will be consistent and tailored to audiences for maximum communications effectiveness. Stakeholder feedback paths to the NRC will also be optimized to ensure that feedback is received, considered, and addressed in a timely manner, as appropriate.

Further, in the area of optimizing the NRC's communications, the near-term strategy is defined as follows:

> Develop and implement a structured, integrated strategy to communicate with internal and external stakeholders having interests in non-LWR technologies.

The contributing activities for this strategy begin in the near term, but will continue throughout the process of the implementing the vision and strategy of all three areas of the readiness for non-LWR activities.

Unlike other strategies described in this report, initial development of the NRC's non-LWR communications strategy document was completed in May 2016 and was designed to address the strategic objective described above. Therefore, this IAP is focused on identifying the supporting actions needed to operationalize and maintain the communications strategy to achieve the strategic objective.

Implementation Action Plan – Strategy No. 6

Contributing Activity No. 6.1: Provide timely, clear, and consistent communication of the NRC's non-LWR requirements, guidance, processes, and other regulatory topics, and provide multiple paths for external feedback to the NRC

The NRC is keeping stakeholders informed of its activities on a periodic basis, and through a variety of methods (tools and platforms). Stakeholders will receive current and timely NRC messaging about non-LWR topics through planned meetings and workshops, and via other periodic communications such as conferences, press releases, and social media channels. Stakeholders will have multiple communication channels available to provide feedback to the NRC.

This purpose of this activity is to specify a coordinated and comprehensive way to distribute NRC's non-LWR messaging, using the communication strategy, to all stakeholders. This includes identifying and deploying the appropriate tools to use for communication with internal and external stakeholders. Broad deployment of the non-LWR strategy will facilitate the NRC's ability to educate all stakeholders on the agency's safety, security, and environmental mission, and enable the NRC to reach out to external stakeholders to discuss challenges for both the agency and industry.

Note that these activities are a subset of the agency's established communication mechanisms established and maintained by OPA. Non-LWR messaging content is developed under Contributing Activity No. 6.2.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Develop, document, and maintain a non-LWR communication strategy for use with internal and external stakeholders	X		NRO, OPA
Make NRC staff aware of the uses and methods available to them to provide non-LWR messaging to stakeholders. Provide periodic refresher briefings.	X		NRO, OPA

Contributing Activity No. 6.2: Develop consistent NRC non-LWR messaging suitable to a range of audiences

The NRC will optimize its communication with non-LWR stakeholders by disseminating clear expectations and requirements for non-LWR regulatory reviews and oversight. These expectations and requirements will be expressed using multiple channels of communication appropriate to different stakeholder interests. NRC messaging will be consistent and tailored to audiences for maximum communications effectiveness. Stakeholder feedback paths to the NRC will also be optimized to ensure that feedback is received, considered, and addressed in a timely manner, as appropriate.

The base set of non-LWR messaging has been provided in the first issue of the non-LWR communication strategy. This activity is to maintain and update the messaging in a methodical manner and to ensure the messaging evolves with the state of the non-LWR industry, non-LWR technical expertise, and NRC non-LWR requirements and guidance.

This contributing activity also includes an educational component to emphasize how staff may best address stakeholder misinformation or misperceptions when the need arises, using accurate standard messaging.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Make NRC staff aware of the approved non-LWR messaging included in the non-LWR communication strategy. Provide annual refresher briefings.	X		NRO, OPA
On a semi-annual basis, review and update messaging in the non-LWR communication strategy Assume 10 updates @ 20 hrs per office, per update	X		NRO, NMSS, RES, NSIR, OPA
Quarterly review with NRO senior executives to discuss non-LWR communications strategy performance against goals	X		NRO, OPA

Contributing Activity No. 6.3: Promote the exchange of non-LWR technical and regulatory experience with the NRC international counterparts and industry organizations

To promote the exchange of non-LWR experiences and expertise, the NRC will engage with new and existing international counterparts. The NRC will participate in various meetings and workshops that focus on operational and regulatory experiences in a multilateral setting. This will afford an opportunity for the NRC to garner experiences from a wide range of counterparts and stakeholders which will enhance NRC's ability to provide it's messaging in a timely fashion, and to receive the latest external information.

This goal of this activity is to ensure the NRC's non-LWR messaging is available for international dissemination as appropriate, and that the messaging is informed by feedback from the non-LWR experience and expertise of our international counterparts.

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
Ensure NRC staff are fully cognizant of and well versed in the NRC's non-LWR messaging (see Contributing Activities 1 and 2) and in any unique requirements for non-	X		NRO, OIP

Supporting Task Description	Job Hours Required	Contract Dollars, $K	Participating Organizations
LWR information exchanges in an international environment			

5.0 SUMMARY OF ESTIMATED RESOURCE NEEDS FOR NEAR-TERM IAP TASKS

(Witheld – For Internal NRC Use Only)

6.0 LIST OF ACRONYMS

Acronyms:

ANS – American Nuclear Society
ASME – American Society of Mechanical Engineers
ASTM – American Society for Testing and Materials
CFR – US Code of Federal Regulations
CNRA - Committee on Nuclear Regulatory Activities
CSNI - Committee on the Safety of Nuclear Installations
CSWG - Codes and Standards Working Group
DICWG – Digital Instrumentation and Controls Working Group
DOE – US Department of Energy
EPRI – Electric Power Research Institute
FTE – Full Time Equivalent
FY – Fiscal Year
GIF – Generation IV International Forum
I&C – Instrumentation and Controls
IAEA – International Atomic Energy Agency
IAP – Implementation Action Plan
IEC – International Electrotechnical Commission
IEEE – Institute of Electrical and Electronics Engineer
INPRO - International Project on Innovative Nuclear Reactors and Fuel Cycles
LWR – Light Water Reactor
MD – Management Directive
MDEP – Multinational Design Evaluation Programme
NMSS – NRC Office of Nuclear Material Safety and Safeguards
NRC – US Nuclear Regulatory Commission
NRO – NRC Office of New Reactors
NRO/DEIA – Office of New Reactors; Division of Engineering, Infrastructure, and Advanced Reactors
NRR – NRC Office of Nuclear Reactor Regulation
OIP – NRC Office of International Programs
OMB – US Office of Management and Budget
PIRT – Phenomenon Identification and Ranking Table
RES – NRC Office of Research
RES/DE/RGGIB – NRC Office of Research, Division of Engineering, Regulatory Guidance and generic Issues Branch
RG – Regulatory Guide
SDO – Standards Development Organization
SRP – Standard Review Plan
VICWG – Vendor Inspection Cooperation Working Group