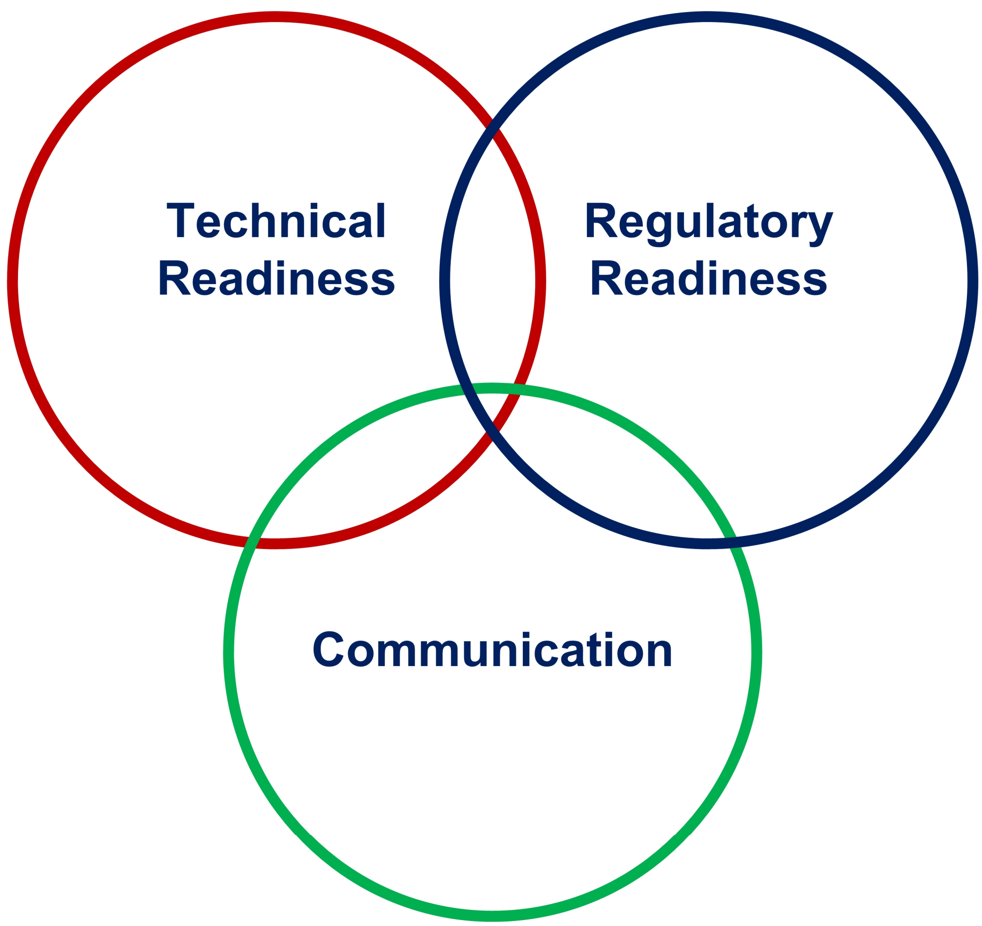
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

ML17319A550 (Final)
Last Revised 12/07/2017

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy – **Strategy 2 Near-Term Implementation Action Plan Progress Report for Fiscal Year 2017**



This page is intentionally blank.

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) is preparing to review and regulate a new generation of non-light water reactors (non-LWRs). As part of these preparations, the NRC developed a vision and strategy document to assure the NRC's readiness to effectively and efficiently conduct its mission for these technologies.

In December 2016, the NRC published the report, "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness" (ADAMS Accession Number ML16356A670). This vision and strategy document describes the objectives, strategies, and contributing activities necessary to achieve non-LWR mission readiness. The NRC prepared implementation action plans (IAPs) to identify the specific activities that the NRC will conduct in the near term (0–5 years), mid-term (5–10 years), and long-term (beyond 10 years) timeframes to achieve non LWR readiness. In 2016, the NRC released its draft near term IAPs to obtain stakeholder feedback (ML16341C620). The staff updated and finalized its IAPs to reflect stakeholder feedback in July of 2017 (ADAMS Accession No. ML17165A069). The near-term IAPs address the following six individual strategies:

1. Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory activities.

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

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

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

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

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

This report provides a summary of progress made toward meeting the objectives and goals for Strategy 2, "Acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews" for the period ending with fiscal year 2017. The NRC's Office of Nuclear Regulatory Research (RES) led the activities under Strategy 2.

During fiscal year 2017, RES focused on familiarization with non-LWR designs, the principal variants being proposed by pre-applicants for regulatory review, and identifying associated information gaps. RES staff, along with staff from the Office of New Reactors (NRO), attended DOE and NRC-sponsored workshops, technology working groups, pre-applicant "drop-in" meetings, and focused training to better understand the reactor systems under development. In

the latter half of the fiscal year, RES began identifying and evaluating computer codes and tools. As a result, most work remains in progress and will continue into fiscal year 2018 using the limited staff hours and program support resources available for this work. To the extent practicable, the agency's strategy relies on leveraging the analytical tools, information, and experiments conducted by other domestic and international partners rather than duplicating this work.

Work completed or nearing completion at the end of fiscal year 2017 includes:

- An initial screening of analysis codes for design basis and beyond design basis event simulation was completed, and a suite of tools for further examination and consideration has been identified. The code suite comprises both NRC-developed and DOE-developed codes. Future efforts will evaluate codes in the code suite against analysis requirements.

- A Phenomena Identification and Ranking Table (PIRT) exercise was conducted for molten salt reactors (MSRs). The PIRT focused attention on fuel salt MSRs due to their novel and unique feature of fuel being part of the coolant. The PIRT is considered preliminary in that design specifics are not available but is useful in that several phenomena requiring simulation could be identified based on existing information.

- A report on Probabilistic Risk Assessment (PRA) was completed. This PRA summarizes previous work and issues for advanced non-LWRs and identifies several policy decisions that may need to be made for non-LWRs.

Work in progress with products expected in fiscal year 2018 includes:

- A contract was awarded to Oak Ridge National Laboratory to examine materials related to advanced non-LWRs systems. Of main concern are the elevated temperatures at which most non-LWRs propose to operate. Corrosion, especially for MSRs, will also be considered. (Materials issues are also being addressed through participation on American Society of Mechanical Engineers [ASME] Standards Committees, which is a Strategy 4 activity.)

- A review of model and simulation needs for the MELCOR code for non-LWRs was initiated. The objective for fiscal year 2018 is to identify models necessary for sodium fast reactors and MSRs.

RES will lead and coordinate the work proposed in this report for FY2018 and FY2019 with NRO and actual work activities will be contingent upon resource availability. Efforts for fiscal year 2018, assuming agreement with NRO and sufficient resources, will include refinement of the analysis code suite and definition of code requirements for each of the major advanced technologies.

1.0 Introduction

1.1 Report Organization

The Implementation Action Plans (IAPs) defined in the vision and strategy document involve the Office of Nuclear Regulatory Research (RES) in several of the strategies. While the majority of the RES accomplishments are associated with the identification and development of computer codes for confirmatory analysis in technical areas such as materials and component integrity, significant work has occurred in nearly all of the six strategies. The following section discusses each strategy and notes the extent of RES involvement. Chapter 2 covers the status of codes and analytical tools, and Chapter 3 provides the status of work by technical area (neutronics, fuel performance, etc.). Chapter 4 summarizes the results accomplished to date and associated conclusions.

1.2 IAP Strategies and RES Involvement

The near-term IAPs address the following six individual strategies:

1. Acquire/develop sufficient knowledge, technical skills, and capacity to perform non-LWR regulatory activities.

   Members of the NRO "core review team" comprised of 1-2 RES staff per technical discipline participated in stakeholder meetings, and at DOE/NRC workshops. DOE's Gateway for Accelerated Innovation in Nuclear (GAIN) working group meetings have been most useful and informative, including separate workshops on modeling and simulation, fuel safety, and thermal-hydraulics. Participation enabled staff to interact with several potential pre-applicants and discuss and assess issues associated with the various non-LWR designs.

   Several RES staff also attended training sessions on molten salt reactors, and one staff member attended a specialized training session on molten salt chemistry. The training sessions and participation in workshops benefited the staff by providing information about the technical and policy aspects of the advanced non-LWR designs.

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

   Strategy 2 has been the principal area of work by RES staff in fiscal year 2017, and involved the technical areas of neutronics, fuel performance, thermal-hydraulics, severe accident phenomena and consequence analysis. Coordinated efforts in these areas have identified computer codes that are potentially suitable for non-LWR confirmatory analysis. Chapter 2 provides details on the Strategy 2 efforts.

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

The Utility-led Licensing Modernization Project, sponsored in part by the Department of Energy, and coordinated by the Nuclear Energy Institute, is preparing white papers on topics such as licensing basis event selection and approaches to the use of PRAs in the design and licensing of advanced reactors. RES supported development of guidance for a flexible, risk-informed review process by participation in several public stakeholder meetings as well as review and comment on these papers.

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

   RES participated in several ANS and ASME Standards Committees in support of Strategy 4 including:

   - *ASME Section III, Division 5 – High-Temperature Materials*
     - *Staff participated on 11 working groups and Subgroups (NRO has the lead for ASME Section III Division 5 and RES is providing technical support.)*

   - *ANS Committees*
     - *Risk-informed Principles and Policy Committee*
     - *Research and Advanced Reactor Consensus Committee*

   - *ANS Working Groups*
     - *ANS 20.1 - Safety Criteria for Fluoride Salt-Cooled High-Temperature Reactors*
     - *ANS 20.2 - Safety Criteria for Liquid-Fuel Molten-Salt Reactors*
     - *ANS 54.1 - Safety Criteria for Liquid-Sodium-Cooled-Reactors*
     - *ANS 30.2 - Categorization and Classification of SSCs for NPPs*

   - *ASME/ANS Joint Committee on Nuclear Risk Management Working Group for non-LWRs (NRO has lead for this activity as coordinated with and supported by RES.)*

5. Identify and resolve technology-inclusive (not specific to a particular non-LWR design or category) policy issues that impact regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs).

   The Office of Nuclear Regulatory Research did not have any activities in support of Strategy 5 in fiscal year 2017.

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

   RES supported many non-LWR outreach activities including the Department of Energy (DOE)/NRC workshops, stakeholder meetings, participation and presentations at workshops and conferences such as the Nuclear Energy Agency's Group on the Safety of Advanced Reactors (GSAR) meetings and the NRC Regulatory Information Conference (RIC).

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

2.1 Strategy 2 Overview

This strategy supports the NRC's strategic objectives of enhancing non-LWR technical readiness and optimizing regulatory readiness by ensuring that the NRC staff has adequate computer models and analytical tools to conduct ~~its~~ reviews of the safety of non-LWR designs in an independent, effective, and efficient manner. RES is developing these codes and tools by conducting research. The research will provide the staff with a basis for new regulations and guidance, and it will enable the staff to assess and confirm the safety of particular designs and fuels.

Although the NRC has historically developed many of the computer codes it uses independently, the approach taken in this report is to leverage the use of established non-LWR codes developed by others where possible and appropriate. The NRC will also leverage its participation in the U.S. Department of Energy's (DOE's) GAIN initiative and other domestic and international initiatives to acquire early insights for new or revised non-LWR codes being developed.

The research efforts for this strategy are categorized into "near-term" efforts that the staff expects to perform over the next five years (fiscal years 2017-2021), "mid-term" efforts that the staff considers likely in the 2022 to 2026 time period, and "long-term" efforts that would be continued as candidate designs mature and are finalized. This report is focused on near-term activities completed in FY2017 and planned for FY2018.

For the purpose of identifying potential nuclear safety codes for this strategy, the staff has considered high temperature gas-cooled reactors, liquid metal fast reactors including sodium fast reactors in section 2.2, and molten salt reactors where the nuclear 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 interactions with prospective applicants and knowledge about these designs and is not intended as a way to "down-select" 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. Because these designs are still in the conceptual or preliminary design stages, the proposed research is intended to be as generic as possible, while focusing on the development of analytical capabilities necessary to ensure reasonable assurance of adequate protection. This can and will be refined as design specifics are made available to the staff, or if particular designs become higher priority candidates for reviews, and licensing or certification.

Individual functional areas have focused contributing activities for this strategy to achieve the strategic objectives. These include: reactor kinetics and criticality; fuel performance; thermal-fluids; severe accidents; and, offsite consequences. It should be noted that most safety issues are multidisciplinary in nature. One functional area often affects several others, and RES emphasized cooperation between staff members working in these various areas as necessary to achieve an effective, efficient research effort.

2.2 Preliminary Code Suite for Design Basis Events

One of the primary objectives of Strategy 2 is the development of codes suitable for confirmatory analysis of high-temperature gas cooled reactors, liquid metal reactors in section 2.1, and molten salt reactors. Codes used by the NRC for confirmatory analysis during the last three decades have been designed and assessed for light water reactors, and are not immediately extendable to these non-LWR designs. Therefore, RES directed initial efforts ~~have been directed~~ at understanding requirements for modeling and simulation of these new designs, and in identifying codes that can meet these requirements. While the development and modification of NRC codes is a potential means to obtaining applicability to non-LWRs, RES has considered codes developed outside of the NRC. In particular, several codes developed under the Department of Energy's (DOE) Consortium of Advanced Simulation of LWRs (CASL) and Nuclear Energy Advanced Modeling and Simulation (NEAMS) programs possess some unique and advanced modeling capabilities that may be adopted for NRC use. RES assessed the code suite for non-LWR confirmatory analysis in 2017, and developed a preliminary plan. The code suite proposed for evaluation is described in Reference 2-1, and includes codes previously developed by the NRC as well as codes developed by DOE.

The proposed code suite for non-LWR confirmatory analysis makes use of existing NRC codes and integrates them with several codes developed through the DOE NEAMS program. The codes in some cases have multiple reactor type applications (i.e., they can be used for more than one design type). Some redundancy is built into the proposed structure to allow for options to meet a currently unspecified pre-applicant schedule.

Figure 1 presents a schematic showing the proposed ~~full~~ suite of NRC and DOE non-LWR codes being considered. The NRC developed codes are shown in gold, while those produced by the DOE are shown in light blue. For each reactor design type, only a subset of codes would be utilized as part of a given calculation. Based on an initial assessment, codes that are expected to play a significant role are the NRC developed or sponsored codes such as TRACE [2-2], FAST [2-3], PARCS [2-4] (and its associated codes for cross sections SCALE [2-5] and gas flow AGREE [2-6]), and codes developed by DOE such as MOOSE [2-7], BISON [2-8], PRONGHORN [2-9], and SAM [2-10]. Should computational fluid dynamics (CFD) analysis be necessary, this would be done using FLUENT [2-11] or possibly the DOE code Nek5000 [2-12]. MELCOR [2-13], while not part of the suite for design basis event analysis, is shown in the figure due to its importance for severe accident and source term evaluation and its usefulness in containment analysis. FAST is a fuel performance code currently under development that combines the capabilities of the FRAPCON [2-14] and FRAPTRAN [2-15] codes.

Reference 2-1 contains descriptions of the individual codes and how the code suite might be applied to various advanced designs. The code suite should be considered as a proposal in that flexibility exists in some of the codes and overlap in capabilities. Future work will be needed to obtain a better understanding of the capabilities of the DOE codes and when overlap in capabilities occurs, what approach is best for the NRC.

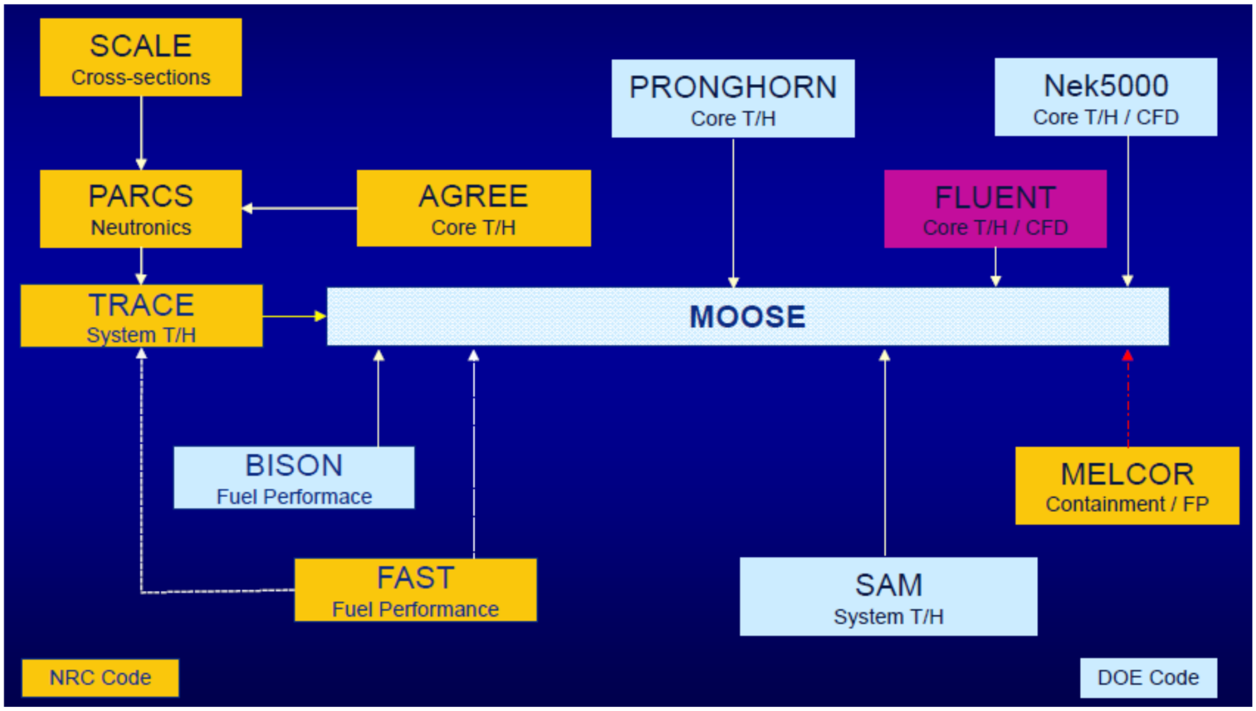


Figure 1. Code suite for reactor safety analysis.

Additionally, in order to ensure that non-LWRs are compliant with nuclear plant siting criteria for the site boundary radiation doses for various design basis accidents (DBA), the Symbolic Nuclear Analysis Package/RADionuclide, Transport, Removal, and Dose Estimation (SNAP/RADTRAD) code will need to be updated for non-LWR DBA. SNAP/RADTRAD [2-16] uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at the exclusion area boundary (EAB) and the low population zone (LPZ) and to assess the occupational radiation doses in the control room (CR) and /or emergency offsite facility for various DBA. SNAP/RADTRAD is expected to be used as a licensing confirmatory code to show compliance with nuclear plant siting criteria at the CR, EAB and LPZ for given DBA scenarios non-LWR accidents.

2.3 Codes for Beyond Design Basis Events and Consequence Analysis

The NRC's MELCOR [2-13] computer code being developed at Sandia National Laboratories is the prime candidate for the evaluation of severe accident progression and source term due to its flexibility and modeling capabilities (i.e., both core and containment phenomena are coupled and modeled together). The code output can be used directly for MELCOR Accident Consequence Code System (MACCS) offsite consequence analysis. Additionally, the MELCOR output source term is directly incorporated into the Radiological Assessment System for Consequence Analysis (RASCAL) emergency response computer code for plume (early) phase dose assessments.

Societal consequences for LWRs are calculated by the NRC, industry, and international organizations using the MACCS [2-17] computer code suite. MACCS models atmospheric

transport and dispersion of airborne radioactive plume segments, offsite protective actions, exposure pathways, health effects, and societal consequences including land contamination, relocated population, and economic cost. MACCS is expected to be used to analyze the consequences of non-LWR accidents.

Under the Federal Response Plan, the NRC is the lead federal agency for all aspects of the federal response with regards to “on site” response actions involving NRC-licensed facilities and sources. The RASCAL [2-18 & 2-19] code is the dose assessment tool used by the NRC, state and local governments and other federal agencies during the plume (early) phase of an incident or event at NRC-licensed facilities. RASCAL rapidly, and with a limited amount of user inputs, generates an atmospheric source term and models its transport and dispersion using real-time meteorological data. RASCAL helps decision makers to develop protective action recommendations (PAR) and implement protective action decisions (PAD) consistent with the EPA Protective Action Guides (PAG). RASCAL is expected to be used during an emergency response to non-LWR accidents.

2.4 Codes for Material and Structural Analysis

Material and structural analyses are performed for LWRs using commercially available codes such as ABAQUS [2-20], ANSYS [2-21], and LS-DYNA [2-22] codes. Probabilistic fracture mechanics analyses are performed for LWRs with the NRC’s xLPR code [2-23] and FAVOR code [2-24] for piping and reactor vessels, respectively. Staff is continuing to assess the gaps in these computational tools for application to high-temperature applications associated with non-LWRs.

2.5 Codes for PRA

Probabilistic risk analysis (PRA) calculations are performed for LWRs using the NRC’s SAPHIRE [2-25] code. The expectation for non-LWRs is that the code will not require modification. The success criteria for PRA is expected to be determined using MELCOR.

2.6 References

[2-1] USNRC, “NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy – Computer Code Suite for Non-LWR Design Basis Analysis,” ML17244A459, 2017.

[2-2] USNRC, “TRACE V5.0 Developmental Assessment Manual,” ADAMS ML120060208, ML120060187, ML120060191, ML120060172, 2008.

[2-3] I.E. Porter, “FAST: The Merger of NRC’s Fuel Performance Codes FRAPCON and FRAPTRAN for Scoping and Regulatory Decision Making”, WRFPM 2017, Jeju, South Korea, September 10-14, 2017.

[2-4] 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).

[2-5] 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).

[2-6] 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.

[2-7] D. Gaston, C. Newman, G. Hansen, and D. Lebrun-Grandie. MOOSE: A parallel computational framework for coupled systems of nonlinear equations. Nucl. Engrg. Design, 239:1768–1778, 2009.

[2-8] R.L. Williamson, J.D. Hales, S.R. Novascone, G. Pastore, D.M. Perez, B.W. Spencer, and R.C. Martineau, “Overview of the BISON multidimensional fuel performance code,” IAEA Technical Meeting: Modeling of Water-Cooled Fuel Including Design-Basis and Severe Accidents, Chengdu, China, October 28–November 1, 2013.

[2-9] Idaho National Laboratory, PRONGHORN Manual, July 2017.

[2-10] R. Hu, SAM Theory Manual, Argonne National Laboratory ANL/NE-17/4, March 2017.

[2-11] ANSYS, Inc., ANSYS FLUENT User's Guide, November 2011.

[2-12] P. Fischer, et al., Nek5000 User Documentation, ANL/MCS-TM-351, 2015.

[2-13] Sandia National Laboratories, “MELCOR Computer Code Manuals, Vol. 2: Reference Manual, Version 2.2.9541,” SAND 2017-0876 O, January 2017 (ADAMS Accession No. ML17040A420).

[2-14] K.J. Geelhood, et al, FRAPCON-4.0: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup, PNNL-19418, Vol. 1 Rev. 2, 2015.

[2-15] K.J. Geelhood, et al, FRAPTRAN-2.0: A Computer Code for the Transient Analysis of Oxide Fuel Rods, PNNL-19400, Vol. 1 Rev. 2, 2016.

[2-16] W.C. Arcieri, D.L. Mylnarczyk, NUREG/CR-7220, “SNAP/RADTRAD 4.0: Description of Models and Methods,” Information Systems Laboratory, Inc., June 2016, (ADAMS Accession No. ML16160A019).

[2-17] D. Chanin, J. Sprung, L. Ritchie, and H.N. Jow, (1990). MELCOR Accident Consequence Code System (MACCS). Volume 1: Users Guide (NUREG/CR-4691 / SAND86-1562). Albuquerque, NM, Sandia National Laboratories.

[2-18] J.V. Ramsdell, G.F Athey, S.A. McGuire, L.K. Brandon, NUREG-1940, "RASCAL 4: Descriptions of Models and Methods," December 2012 (ADAMS Accession No. ML13031A448).

[2-19] J.V. Ramsdell, G.F Athey, J.P. Rishel, NUREG-1940 Supplement 1, "RASCAL 4.3: Descriptions of Models and Methods," May 2015 (ADAMS Accession No. ML15132A119).

[2-20] ABAQUS 2016 Online Documentation, Dassault Systemes, 2015, http://50.16.225.63/v2016/.

[2-21] ANSYS User Manual, ANSYS, Inc., 2017.

[2-22] LS-DYNA R10.0 Manual, Livermore Software Technology Corporation (LSTC), 2017, http://www.dynasupport.com/manuals.

[2-23] xLPR v2.0 Theory Manual, U.S. Nuclear Regulatory Commission, 2017.

[2-24] FAVOR v16.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, P.T. Williams, T.L. Dickson, B.R. Bass, and H.B. Klasky, September, 2016, ORNL/LTR-2016/309, ADAMS ML16273A033.

[2-25] Idaho National Laboratory, Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) Version 8 (NUREG/CR-7039), June 2011.

3.0 Progress Summary by Functional Area

3.1 Overview

Functional areas are organized based on where emphasis has been categorized for conventional light water reactors. These are reactor kinetics and criticality, fuel performance, thermal-fluid phenomena, severe accident phenomena, offsite consequence analysis, materials and component integrity, and probabilistic risk assessment. These areas are not necessarily all-inclusive, and new areas may arise or be adjusted as the non-LWR designs become more specific and as NRC safety evaluations proceed.

No functional area is independent, and each depends on others; thus, the safety issues and the associated review 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, applicants, and other stakeholders to assess safety and risk realistically without being unnecessarily conservative.

3.2 Functional Area: Reactor Kinetics and Criticality

3.2.1 Overview

In order to support licensing reviews, confirmatory calculations that model reactor kinetics and criticality phenomena will aid staff in their evaluation of advanced reactor concepts. Reactor kinetics and criticality refer to the relationship between the neutron population and the reactor power within the core. The accurate modeling of kinetics and criticality are crucial to the understanding of the core behavior in all phases of operation and during steady state and transient conditions. Specifically, the amount of core thermal power being produced is directly proportional to the net number of neutrons that are produced from fissions (after accounting for other neutrons being absorbed in the fuel and structure). Accurately modeling neutronic phenomena will be important for non-LWR technologies being considered in that some of the designs call for more spatial detail (TRISO fuel), moving fuel (liquid MSR), or more energy detail (SFRs). The purpose of the IAP is to prepare modeling and simulation (M&S) capabilities ~~within RES~~ to be able to support licensing of non-LWRs. The other objective is to validate these M&S tools against publicly available data sources. Broadly, this includes molten salt reactors (both liquid and solid fueled forms), fast reactors (metallic fuel, oxide fuel, sodium cooled, liquid metal cooled, etc.), and HTGRs (high temperature gas cooled reactors that include TRISO fuel). There are nuclear vendors interested in all three reactor types.

The role of RES is to support licensing with independent confirmatory safety analysis. This results in the development of codes, data, and expertise that support staff in their understanding of the methodologies being employed by licensees. Research programs also help to establish sources of uncertainty and establish safety margins under varying plant conditions. Faced with a dynamic external environment, with the possibility that RES will face more budget constraints, it is a goal to build on past NRC and DOE research where feasible. This will enable RES to leverage past government investments. For HTGRs, RES intends to leverage past work that

has already been completed the next generation nuclear plant (NGNP) program. The NGNP was a DOE project to use a Very High Temperature Reactor (VHTR) to drive process heat applications, and the US NRC had research programs to develop an Evaluation Model (EM) for a tentative license application. As part of this research, the TRIPEN/AGREE codes have been tested against the first of the loss-of-forced cooling (LOFC) series of tests of the high temperature test reactor (HTTR) in Japan. Versions of TRACE/PARCS have been applied to the analysis of fast reactors systems (sodium and gas cooled) in the framework of the Gen-IV forum within Europe. For both of these reactor classes, in FY2018 and FY2019 RES will be able to upgrade and validate our existing reactor physics toolset (e.g., SCALE, GenPMAXS, PARCS) for potential, high priority designs. In FY2018, TRIPEN/AGREE will continue to be accessed for the HTTR-LOFC. An option that will be considered is the use of PRONGHORN for gas-cooled reactor analysis. PRONGHORN was developed initially for pebble bed cores, and may have an advantage in its application and assessment. In FY2018, TRIPEN/AGREE will continue to be accessed for the HTTR-LOFC. In FY2018, SCALE and PARCS will start to be upgraded for SFR analysis. For molten salt reactors with liquid fuel, algorithms will be developed to accommodate the drift of delayed precursors in the primary loop external to the core. As a result of leveraging past and ongoing NRC and DOE research, and through research programs available to RES through our international partners, RES expects to proactively prepare to support the agency in the review of future non-LWR technologies. RES will need to leverage collaboration with domestic and international non-LWR stakeholders in order to develop this code suite.

3.2.2 Work Completed in FY2017

The purpose of IAP strategy 2 is to prepare modeling and simulation (M&S) capabilities to support the NRC’s review of a design certification or a construction permit/operating license application from the class of reactors that have been binned as non-LWR. With respect to reactor physics and criticality (core power as a result of neutron fissions), this is necessarily closely tied to other conditions in the reactor such as temperature, flow, and pressure. It is RES’s role to develop independent tools that can model these phenomena with sufficient resolution to support regulatory decision making. Before RES can select or upgrade a code suite for a specific non-LWR type, it is necessary to first understand the scenarios (plant transients and normal operation) that the agency will need to assess to confirm safety ~~simulate~~. In order to model the scenarios, RES needs to understand sufficiently the underlying phenomena and physical processes that characterize these plant conditions.

The NRC has the least experience with liquid metal fast spectrum and molten salt (fast and thermal spectrum, with solid or liquid fuel) designs. The NRC has never licensed a MSR design. Thus, RES has prioritized code development for these general reactor classes. In FY2017, RES started the process of understanding the modeling needs with respect to MSRs. RES staff and contractors conducted to a literature survey of past and proposed MSR designs to prepare for the PIRT. As there wasn’t enough information during this expert elicitation to characterize or rank the expected phenomena, this exercise was characterized as a pre-PIRT.

RES hosted the pre-PIRT for MSRs to identify figures of merit (FOMs) and the phenomena relevant to both neutronic and thermal hydraulic analysis with fast and thermal spectrum systems. Panelists included experts from several national laboratories, universities, and RES staff. To prepare for the meeting, Brookhaven National Laboratory (BNL), serving as the

primary RES partner ~~contractor~~, developed a primer report that gave an overview of MSR technology for thermal-liquid fueled fluoride systems, fast molten chloride systems, and solid fuel FHRs. This report outlined simulation scenarios (e.g., temperature changes, reactivity changes, loop breaks, loss of forced flow, etc.) for normal operation and licensing-basis events in both liquid-fueled and solid-fueled MSRs. The report also highlighted important physical processes in MSRs that are important for these simulation scenarios. In addition to this report, the pre-PIRT panelists prepared for the meeting by reviewing historical technical reports that were developed in support of the Molten Salt Reactor Experiment (MSRE) at ORNL, along with recent ANS summaries documenting some of the current research in the design and research communities.

The pre-PIRT deliberations spanned several days. Panelists identified six Figures of Merit (FOMs) for important for thermal-hydraulic simulation and five FOMs for neutronic simulation of MSRs in both fast and thermal systems. Panelists also identified and described many of the neutronic and thermal-hydraulic phenomena relevant to the FOMs, and these were documented during pre-PIRT proceedings. Subsequent to the pre-PIRT deliberations, several of the panelists produced follow-on reports to document the FOMs and expand on the phenomena important to the modeling and simulation of MSRs. These were documented and appended to the draft Brookhaven report [3.2-23] that, in addition to the proposed simulation scenarios, expanded on the important phenomena that were discussed during pre-PIRT deliberations.

For SFRs, the NRC obtained the code changes that the Paul Scherrer Institute (PSI, in Switzerland) made for TRACE/PARCS for fast neutron spectrum designs. This collaboration was enabled through the CAMP (Code Application and Maintenance Program). For PARCS, these changes included a fuel expansion reactivity feedback model, along with a methodology to logarithmically interpolate fuel temperature cross sections. PSI made these changes to PARCS version v261m00, and the code changes were documented in an internal PSI report [3.2-24]. Programmatically, a series of tasks was developed in order to undertake fast reactor development, documentation, and testing. These tasks are expected to start in FY2018 and end in FY2019.

For HTGRs, progress has been made with modeling the HTTR-LOFC transient with TRIPEN/AGREE. Contractors have continued to parameterize model benchmark inputs such as graphite thermal conductivity, fuel compact specific heat, and the flow present in the auxiliary cooling system with the result that the timing of the post-transient power rise is more accurately predicted. For SCALE, improvements have been made to the source code of the resonance processing modules, and code optimizations have been made to the neutron transport modules as well. The result is substantial gains in code speed for lattice physics calculations. This is particularly relevant for HTGRs in that the transport of neutrons through TRISO coated fuels residing in graphite matrix is computationally challenging. This has resulted in an improvement to RES's toolkit for modeling the double heterogeneous fuel arrangements present in HTGRs (and also for the TRISO fuel kernels present in solid fuel MSRs). In addition, the Shift Monte Carlo code was integrated into the SCALE TRITON sequence to provide generalized geometry capabilities with future applicability to the complex heterogeneous configurations presented by several non-LWR designs.

3.2.3 Work in Progress

RES staff is planning a code suite and methodology for MSRs that adequately captures the neutronic phenomena documented in the pre-PIRT exercise. It is expected that this will be revised as more information becomes available and as RES develops expertise in this area. RES received the draft pre-PIRT report from the contractor with main responsibility for leading the pre-PIRT, coordinating the deliberations, and integrating all input. RES staff are reviewing the report, and the report will be finalized in the near-term.

For SFRs, a series of tasks is being developed in order to undertake fast reactor development, documentation, and testing.

3.2.4 Work Recommended for FY2018 & FY2019

Customarily reactor analysis is undertaken with a two-step methodology. During the first step, transport calculations are pre-computed for the different fuel types at an expected set of core thermal-hydraulic states and at different sets of depletions occurring along different histories. During the second step of reactor analysis, a diffusion code interpolates and extrapolates these pre-computed cross sections according to a pre-determined methodology. This methodology is determined beforehand such that the cross sections are functionally combined to adequately characterize core multiplication. In sodium cooled fast reactor systems, fast neutrons induce a significant amount of fissions before being thermalized (“slowed down”). These fast neutrons are typically streaming (leaking) from the core on a larger scale than what is occurring in LWRs. For this reason, the ordinary geometric distortions that occur with core burnup and fluence have a larger effect on the neutron population than what would be important for LWR analysis. In fast reactor analysis it is thus necessary to adjust the parameterization of the reactivity coefficients to account for the significant feedback due to fuel displacement in the radial and axial directions in addition to the Doppler and coolant density feedback.

In FY2018 and FY2019, the original PSI fast reactor code changes will be pulled in to the PARCS source and changed to be consistent with the most recent official PARCS version. Testing of these code changes in a regression (code-to-code) and assessment (code-to-data) sense will also proceed in parallel. The first part of this work will consist of developing a fast reactor cross section methodology for U-10Zr fuel alloy (i.e., the development of a way to parameterize the cross sections in terms of important state parameters), along with developing a methodology to incorporate the cross section branchings into the PARCS system. Comparisons of the upgraded PARCS will be made to steady-state Monte Carlo results, along with assessment of TRACE/PARCS transient simulations against publicly available data and publications that may be available from EBR-II and Phenix. The final product will contain documentation of the code features in a user manual, documentation of the code changes in a programmer’s manual, and documentation of the assessment against fast reactor data and benchmarks. These code development and assessment activities will serve as a surrogate for a gap analysis study for PARCS as applied to SFRs. Potential development and assessment will be documented and isolated as a series of “open items” that should be addressed, either through the development of additional nuclear analysis methods, or the incorporation of standard SFR nuclear reactor methodologies into PARCS. In addition, ORNL will conduct a gap analysis study for modifying the SCALE code system for SFR calculations. This will build upon previous work that involved the development a fine multigroup cross section library for SCALE.

For HTGRs, RES will continue to evaluate TRIPEN/AGREE and PRONGHORN for their relative strengths in addressing parameters important for nuclear safety. A review of these codes will also be documented in a gap analysis report. The report will describe the relative merits of the codes in terms of the soundness of the underlying physics and numeric approximations, code usability, current assessment and testing, and code documentation.

For MSRs, a similar process will be followed for selecting a code suite, with consideration for all of the DOE-supported research at national laboratories and domestic universities. The expected result of the process will be a gap analysis detailing the most appropriate methodology-code suite that NRC/RES should use going forward, with the same considerations for the scientific approach, code usability, and code assessment.

3.2.5 References

[3.2-23] Phenomena Important in Molten Salt Reactor Simulations. BNL-XXXXXX-2017-IR, Draft September 28, 2017, David Diamond. Brookhaven National Laboratory. [ML17279A070]

[3.2-24] Enhancement of the PARCS, TRAC/AAA, and ERANOS Codes for Use within the FAST Code System. TM-41-05-21, October 7, 2005. Paul Scherrer Institut. [ML17279A099]

3.3 Functional Area: Fuel Performance

3.3.1 Overview

Fuel performance phenomena refer to the thermo-mechanical response of the nuclear fuel in response to operating conditions and transient scenarios. Work in the area of computational capability development for fuel performance of advanced reactors is limited at this time due to the preliminary nature of fuel designs intended for non-LWR reactors. Initial efforts will focus on increasing the knowledge level of in-house staff and making existing fuel performance computational tools more flexible and modular in general. This will give the NRC staff expertise needed for selection of appropriate tools and computational approaches for future fuel analysis and code development once fuel designs for non-LWRs mature and stabilize.

3.3.2 Work Completed in FY2017

In 2017, the staff effort included outlining the objectives and tasks of a contract to decouple the physics from the numerical solution in the Fuel Analysis under Steady-state and Transients (FAST) code, NRC's fuel performance code designed for LWR Zr-UO_2 fuel. The staff effort also included in-house efforts to implement a more modular computational approach in FAST, whereby all or most material properties are stored in a Material Library (MatLib) rather than being embedded directly into the physics models solved within the code. This creates flexibility in using the code to assess new materials. Staff effort this year included beginning to add material properties and correlations relevant to advanced reactor fuels into MatLib. Specifically, helium and liquid sodium properties have been added to MatLib. Finally, staff efforts have included information gathering for a summary report on fuel for non-LWRs. Staff met with

experts with experience with historic non-LWR research and development and conducted interviews to collect information about historic advanced reactor fuel R&D, advanced reactor fuel operating experience, available codes for fuel performance analysis and their status.

3.3.3 Work in Progress

Work is continuing in each of the areas identified above. The contract for work to decouple the physics from the numerical solution in FAST has begun. The scope identified in the contract is expected to continue for approximately two years once started. Work will continue in-house to improve the functionality and content of MatLib to achieve optimal flexibility to address the unique characteristics of new materials. Staff will continue to expand MatLib to include more material properties and correlations relevant to advanced reactor fuels. One area of focus will be adding metallic fuel (e.g., U10Zr) properties to MatLib. Finally, staff will write a report, consolidating collected information on historic advanced reactor fuel R&D, advanced reactor fuel operating experience, available codes for fuel performance analysis and their status.

3.3.4 Work Recommended for FY2018 & FY2019

Work to develop the computational capability for fuel performance of advanced reactors is a complex endeavor that will span multiple years. It will need to be informed by, and updated for, fuel designs as non-LWR designs mature and stabilize. Following on from the addition of metallic fuel properties in MatLib, staff will review available literature on metallic fuels and determine if these fuel designs will require the modeling of additional fuel performance phenomena beyond those relevant for the current operating reactor UO2/Zirconium-based fuel designs, such as fuel creep. If so, work may be completed to include a fuel creep model in FAST.

More generally, in FY 2018 and FY 2019, RES staff will continue to review historic non-LWR fuel performance codes and evaluate their readiness for supporting licensing review of future non-LWR fuel by focusing the review on existing code validation databases. Many of the historic codes have not been continually maintained and there may be a need for further code development before they can be used for future licensing work. Codes that could be reviewed include the PARFUME and MELCOR codes (developed by Idaho National Lab and Sandia National Lab, respectively), which model the performance of TRISO fuel, the fuel form expected to be used in gas-cooled advanced reactors. The PARFUME code is designed to predict the behavior of TRISO particle fuel during reactor normal operation and heat up accidents, and the MELCOR code includes preliminary models for core-wide fission product release from TRISO fuels. For sodium cooled reactors, fuel performance codes include LIFE-METAL and FPIN2. The LIFE-METAL computer code is the analytical tool developed at Argonne National Lab (ANL) to model the response of metal fuel and blanket elements to steady-state and operational transient conditions. The FPIN2 code, also developed by ANL, 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 two additional codes developed by ANL; the STARS code for steady-state initialization and FRAS3 for transient fission gas behavior. The SASSYS code is a whole-core response code developed by ANL that includes a less detailed model for the fuel element thermal-mechanical response. Staff could review these codes, determine their state of validation, verification and assessment, evaluate their potential application for NRC

licensing purposes, and document the needs for further code development in FY 2018 and FY 2019.

It may also be possible by FY 2018 and FY 2019 to identify potential shortcomings in the existing fuel performance codes and formulate a more detailed plan for future code development needs on a technology-specific basis.

3.3.5 References

None.

3.4 Functional Area: Thermal-Fluid Phenomena

3.4.1 Overview

Thermal-fluid 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 as 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 because a wide variety of fluids is proposed for use in non-LWR designs.

The coolant used in the primary system is significant in that its selection simplifies (or complicates) the analysis. Water is somewhat unique and complex in that most systems allow it to boil, and two-phase flow becomes an important part of the evaluation as it often leads to large uncertainties due to its nature. Single-phase fluids (such as with helium-cooled systems or in other designs where boiling is highly unlikely) are generally easier to deal with. 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.

In 2017, efforts were focused on molten salt and sodium fast reactor issues. Molten salt reactors received initial emphasis because of a shortage of information on physical processes and the relative uniqueness of MSRs. Obtaining information on and, in some cases, testing new thermal-fluids codes developed outside of the NRC also received significant attention.

3.4.2 Work Completed in FY2017

Two “pre-PIRT” exercises were conducted in 2017 with the intent of identifying major thermal-hydraulic phenomena and accident scenarios of interest to molten salt reactors and sodium fast reactors. In a traditional PIRT, a panel of experts reviews a near-final design and its safety systems and, for a selected accident scenario, identifies the most dominant physical processes. These non-LWR exercises were termed “pre-PIRT” in that neither the design specifics nor the accident scenarios were available to the panel of experts. The panel reviewed available information on several designs and in a generic fashion identified physical processes expected

to be important. Accident scenarios considered in past molten salt and sodium fast reactor designs were used to guide Figures of Merit on which the PIRT phenomena are ranked. Although the "pre-PIRT" phenomena are expected to provide a rationale for design basis event code selection, it is expected that a full PIRT will be conducted once a design is submitted to the NRC for design certification review, and the safety systems are known. This second, full-PIRT exercise should identify any remaining important phenomena and modeling needs as well as phenomena associated with decay heat removal systems.

The "pre-PIRT" exercise for molten salt reactors was completed and identified modeling and simulation needs for two types of MSRs: a fluoride fuel salt reactor with a thermal spectrum and a chloride fuel salt reactor with a fast spectrum. The report produced by the PIRT panel [3-1] discusses the phenomena for each general type and gaps in the experimental database.

3.4.3 Work in Progress

Identification of important phenomena and of potentially important accident scenarios requiring new code features was an initial undertaking in 2017, and that work will continue in 2018. In 2017, that initial work was directed toward molten salt reactors as those designs appeared to be the largest uncertainty in the analytical needs. For sodium fast reactors, a "pre-PIRT" was formed, and a report discussing important phenomena and accident scenarios is still under preparation. A report on sodium fast reactors is expected to be completed early in fiscal year 2018, and will draw heavily on existing technical information available to the staff. (No efforts are anticipated in 2018 on gas-cooled reactors because of the PIRTs developed as part of next generation nuclear plant (NGNP) research.

The proposed code suite [3-2] described in Section 2 contains flexibility, and some work is necessary to make an informed decision on codes where more than one code has applicability. Two projects were initiated in late 2017 to assist in code selection:

- An agreement was placed with Idaho National Laboratory to perform code assessment for the PRONGHORN code. Previously, PRONGHORN was found to have good agreement with integral test data applicable to gas-cooled reactors. These assessment cases are being repeated because of substantial revisions to PRONGHORN.

- An agreement was placed with Argonne National Laboratory to obtain assistance with the SAM code for sodium fast reactor analysis. Evaluations of SAM are being performed in-house with NRC staff. However, RES has requested a modest amount of technical support from the original developers to assist with code and/or input model problems and for knowledge transfer.

3.4.4 Work Recommended for FY2018 & FY2019

Three efforts are recommended for 2018-2019 time period pending resources. Each of the following efforts are intended to identify and resolve "gaps" in current capabilities:

- As the PIRTs are completed, the codes in the code suite should be reviewed and deficiencies in models and correlations or in modeling features should be identified. Part of this effort may require development of conceptual analysis models for major non-LWR

design types. This gap identification on code capabilities is expected to enable better estimation of resource needs in 2019 and beyond. This effort should include testing of any DOE codes to ensure that they are capable of what is claimed and that they can be executed from NRC headquarters on required high-performance computing platforms or on internal NRC systems.

- Code assessment is considered a critical element of code development and finalization of any code selections that must be made. For each major design type (gas-cooled pebble bed, gas-cooled prismatic, sodium fast, fluoride thermal spectrum molten fuel salt, chloride fast molten fuel salt), a set of primary assessment tests should be identified and data obtained for use with code assessment.

- Gaps in the experimental database should be identified and explained to interested stakeholders. While closely associated with Task B, the objective here is to identify data that are missing from the existing database and to emphasize to stakeholders that new experimental testing may be necessary to support code development and assessment.

3.4.5 References

[3.4-1] USNRC, "Physical Phenomena and Issues Associated with Molten Salt Reactors" (in preparation), November 2017.

[3.4-2] USNRC, "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy – Computer Code Suite for Non-LWR Design Basis Analysis," September 2017, ADAMS ML17244A459.

3.5 Functional Area: Severe Accident Phenomena

3.5.1 Overview

Work in severe accident phenomena is intended to identify computational tools for analysis and new phenomena to be considered for various designs. Code modeling will rely on PIRTs. PIRTs have been developed for gas-cooled reactors as part of NGNP [3.5-1 – 3.5-3] as well as some preliminary work for SFRs [3.5-4 – 3.5-9]. The MELCOR computer code [3.5-10 – 3.511] has been developed and is being refined at Sandia National Laboratories. MELCOR is the prime candidate for the evaluation of severe accident progression and source term due to its flexibility and modeling capabilities (both core and containment phenomena are coupled and modeled together). In addition, the code output can be used directly for MACCS offsite consequence analysis. The code is a fully integrated, engineering-level computer code designed to model the progression of postulated accidents in both reactors and in non-reactor systems (e.g., spent fuel pool). For non-light water reactor designs, current capabilities exist for gas-cooled reactors and models for sodium fires, and efforts are being made to use the code for prediction of source term for these designs.

3.5.2 Work Completed in FY2017

The work scope for FY2017 included the following:

- Review severe accident work performed for NGNP and confirm that development planned for MELCOR should continue. Identify major development efforts to ensure MELCOR is capable of GCR confirmatory analysis.

- Review the capabilities of MELCOR for SFR confirmatory analysis for severe accidents. Select a code for future development and identify models most necessary for the development efforts.

- Review the capabilities of MELCOR for molten salt reactor (MSR) analysis. Propose a preliminary plan for severe accident analysis of fixed core MSRs and liquid fuel MSRs.

<u>HTGR/GCR</u>

A review of the existing modeling capabilities for application to HTGR/GCR designs documented in MELCOR code manuals began in July 2017 by the code development team. The major code models reviewed included (1) PBR and PMR modifications to the core package (e.g., correlations for effective heat transfer in a packed pebble bed and effective heat transfer through graphite blocks), (2) fuel failure and fission product release from PBR pebbles, PMR compacts, (3) fission product and dust transport (i.e., turbulent deposition and lift-off/resuspension), (4) balance-of-plant performance (RCCS, helium turbomachinery, heat exchangers), (5) stratified flow for air ingress, and (6) point kinetics model. Additionally, the strategy for analyzing an accident was reviewed (e.g., PBR400 PLOFC and assessment of radionuclide release from TRISO fuel particles).

<u>SFR</u>

A review of the CONTAIN-LMR models [3.4-2] for containment thermal-hydraulic analysis began in July of FY2017. A few aspects of sodium chemistry modeling related to pool fires, spray fires, and aerosol reactions have been integrated in MELCOR from CONTAIN-LMR for the modeling of containment response to coolant leaks. A generalized modeling approach exists in MELCOR for the modeling of any working fluid's equation of state (EOS). Accordingly, the SIMMER code EOS models have been implemented and benchmarked in MELCOR for the modeling of RCS and containment systems. SIMMER code was developed at the Los Alamos National Laboratory (LANL) under NRC/DOE sponsorship. If needed, appropriately simplified models for fuel failure and fission product release to coolant may be considered in the modeling of primary system response. Some specific phenomenology being considered for inclusion in MELCOR are ex-vessel sodium/concrete interaction, hot gas layer phenomena and aerosol entrainment effects accompanying sodium fires.

<u>MSR</u>

A review of MSR technical references has begun. Several potentially useful references are compiled in a spreadsheet obtained from INL's GAIN initiative. In addition, NRC staff attended the MSR workshop at Oak Ridge National Laboratory in April 2017. MELCOR capabilities exist for implementing molten fluoride/chloride salt EOS and transport properties.

3.5.3 Work in Progress

The focus for HTGR/GCR is on the review of the source code architecture for application to current design concepts and potential new model developments. The MELCOR code documentation including the Reference Manual [3.5-11] and User's Guide [3.5-10] is updated when necessary. Testing of the existing SFR models involves comparison of the MELCOR predictions to available experiments and code-to-code comparison. In addition, new models are integrated into MELCOR (sodium spray/pool fire models and atmospheric chemistry) based on the available CONTAIN-LMR modeling approach. The MELCOR code development team is exploring the possibility of modifying HTGR models to applicable MSR reactors, since several MSR designs use TRISO fuel and HTGR core components (e.g., graphite).

3.5.4 Work Recommended for FY2018 and FY2019

Extensive modeling enhancements were made in MELCOR in 2008-2012 to support NRC's review of NGNP for application to HTGR/GCR accident analyses; however, the work was stopped in 2012 when the program was suspended before all planned development was completed and before all models were fully assessed. Additional work in FY 2018 and FY 2019 will focus on (1) code assessments and accident analysis for both PBR and PMR, and (2) review of HTGR models for potential enhancement of existing models and/or implementation of new models based on extensive testing and evolution of design concepts, including:

- Point-kinetics model
- Balance-of-plant components (e.g., integral heat exchangers)
- Behavior of fission products and graphite dust (release from the fuel, transport to the coolant, and deposition in the primary system)
- Reactor cavity cooling system (RCCS)

Significant progress has been made in implementing SFR-specific capabilities into MELCOR. SFR-specific activities needed for severe accident and source term analysis include further integration of CONTAIN-LMR into MELCOR (ex-vessel phenomena including sodium/concrete interaction), and enhancing existing models and potentially implementing new models based on extensive testing and evolution of design concepts. In particular, the work will focus on the review of SAS4A models for possible coupling or simplified implementation in MELCOR. Additional modeling requirements include fuel failure and fission product release models, and extension of core components (e.g., provision for metal fuel and steel cladding). Assessment is an integral part of MELCOR code improvements and includes cross-comparison of MELCOR with IRSN's ASTEC-Na computer code developed in France for sodium fire experiments as well as MELCOR comparison to IRSN and available domestic experiments.

There are existing capabilities in MELCOR that were developed for HTGR and SFR accident analysis that can be adapted for modeling MSRs. The work envisioned in FY 2018 and FY 2019 involves exploring the existing code architecture and development of a modeling approach for differing types of MSR designs (liquid fuel and solid fuel) as well as adapting existing generic working fluid capabilities in MELCOR for use with different MSR salts.

An initial MELCOR assessment will be documented in a gap analysis report, and MELCOR modernization efforts will be evaluated for efficient code development and maintenance and application to emerging technologies.

3.5.5 References

[3.5-1] "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Volume 1: Main Report," NUREG/CR-6944, Vol. 1 (March 2008).

[3.5-2] "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Volume 3: Fission-Product Transport and Dose PIRTs," NUREG/CR-6944, Vol. 3 (March 2008).

[3.5-3] "TRISO-Coated Particle Fuel Phenomena Identification and Ranking Tables (PIRTs) for Fission-Product Transport Due to Manufacturing, Operations, and Accidents," NUREG/CR-6844, Vol. 3 (July 2004).

[3.5-4] D.A. Powers et al., "Advanced Sodium Fast Reactor Accident Source Terms: Research Needs," SAND2010-5506 (September 2010).

[3.5-5] M. Denman et al., "Sodium Fast Reactor Safety and Licensing Research Plan – Volume I," SAND2012-4260 (May 2012).

[3.5-6] M. Denman et al., "Sodium Fast Reactor Safety and Licensing Research Plan – Volume II," SAND2012-4259 (May 2012).

[3.5-7] T. H. Fanning, A. J. Brunett, and T. Sumner, eds., "The SAS4A/SASSYS-1 Safety Analysis Code System," ANL/NE-16/19, Nuclear Engineering Division, Argonne National Laboratory (March 2017).

[3.5-8] "Regulatory Technology Development Plan: Sodium Fast Reactor, Mechanistic Source Term Development," ANL-ART-3, Nuclear Engineering Division, Argonne National Laboratory (February, 2015).

[3.5-9] Regulatory Technology Development Plan: Sodium Fast Reactor, Metal Fuel Radionuclide Release, ANL-ART-38, Nuclear Engineering Division, Argonne National Laboratory (February 2016).

[3.5-10] "MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.2.9541," SAND 2017-0455 O, Sandia National Laboratories, January 2017 (ADAMS Accession No. ML17040A429).

[3.5-11] "MELCOR Computer Code Manuals, Vol. 2: Reference Manual, Version 2.2.9541," SAND 2017-0876 O, Sandia National Laboratories, January 2017 (ADAMS Accession No. ML17040A420).

[3.5-12] L.L. Humphries and D. Louie, " MELCOR/CONTAIN LMR Implementation Report – FY16 Progress," SAND2016-12101 (November 2016).

3.6 Functional Area: Offsite Consequence Analysis

3.6.1 Overview

The objective of the offsite consequence analysis functional area is to ensure that modeling capabilities in the MELCOR Accident Consequence Code System (MACCS) computer code suite are updated as needed to model offsite consequences for radiation releases from non-LWR designs. MACCS is used by the NRC, industry, and international organizations, and it models atmospheric transport and dispersion of airborne radioactive plume segments, offsite protective actions, exposure pathways, dosimetry, health effects, and societal consequences including land contamination, relocated population, and economic cost. MACCS has traditionally been used by the NRC for analysis of releases from LWRs, but a variety of differences may warrant modeling updates for non-LWRs including (1) radionuclides including their chemical, physical, and biological properties; (2) environmental release pathways; (3) atmospheric transport and deposition; (4) emergency response; and (5) non-radiological hazards. In addition, this effort evaluates and improves MACCS's capability to perform probabilistic calculations of offsite dose as a function of distance. Such calculations could be used to inform emergency planning zone (EPZ) size determinations when estimating the distance at which EPA Protective Action Guide (PAG) [3.6-1] dose levels would be exceeded.

This objective of the offsite consequence analysis functional area also ensures that the dose assessment modeling capabilities for both design-basis accident in the SNAP/RADTRAD code and the severe accident in the RASCAL emergency response computer code are updated as needed for non-LWR designs.

- SNAP/RADTRAD is used by the NRC, industry, and international organizations as a licensing confirmatory code to evaluate compliance with nuclear plant siting criteria and to assess the occupational radiation doses in the control room and/or emergency offsite facility for various DBAs. SNAP/RADTRAD has traditionally been used by the NRC for analysis of licensing actions from LWRs, but a variety of differences may warrant modeling updates for non-LWRs including (1) radionuclides including their chemical, physical, and biological properties; (2) environmental release pathways; and (3) atmospheric transport and deposition.

- RASCAL is the dose assessment tool used by the NRC, state and local governments, industry, and other federal agencies during the plume (early) phase of an incident or event at NRC-licensed facilities. RASCAL has traditionally been used by the NRC for analysis of releases from LWRs, but a variety of differences may warrant modeling updates for non-LWRs including (1) radionuclides including their chemical, physical, and biological properties; (2) environmental release pathways; and (3) atmospheric transport and deposition. Additionally, updates to the models in RASCAL for non-LWR designs would aid emergency response decision-makers in the development of PAR and implement PAD ensuring compliance with the EPA PAG [3.6-1] dose levels during the plume (early) phase of an event or incident.

3.6.2 Work Completed in FY2017

Work completed to date is the development of a statement of work and establishing the contract vehicle to complete the work. The Sandia National Laboratories was selected as the partner for expertise in developing and maintaining the MACCS code, and the FY2017 funds were obligated to begin the work.

3.6.3 Work in Progress

Two tasks are currently in progress. The first task directs Sandia to review the literature on non-LWR accident characteristics to identify and prioritize changes to the MACCS code suite that would be needed to appropriately model non-LWR offsite consequences. The second task directs Sandia to evaluate the capability of MACCS for probabilistic calculations of offsite dose as a function of distance to inform EPZ size determinations. A predecessor computer code to MACCS was used in NUREG-0396 [3.6-2] to support the technical basis for EPZ sizes for large LWRs. NRC staff are in progress of recreating such calculations using the current version of MACCS (v3.10) to identify what modeling assumptions are needed and whether code changes are necessary.

3.6.4 Work Recommended for FY2018 & FY2019

The two tasks mentioned above in Section 3.6.3 will continue into FY2018. As these tasks identify and prioritize modeling changes that are needed to appropriately model offsite consequences from non-LWRs, the modeling changes will be implemented in separate tasks. These separate tasks also direct the contractor to provide complete documentation and verification of modeling changes implemented. In addition, planned work will assess the underlying architecture of the MACCS code to ensure that it remains modern and flexible for modeling of non-LWR accident releases well into the future.

In FY-2019 it is anticipated that RES will begin assessing (1) radionuclides that could be released in a severe accident including their chemical, physical, and biological properties; (2) environmental release pathways unique to advanced reactors; and (3) any atmospheric transport and deposition unique to advanced reactors with respect to their implementation in the SNAP/RADTRAD and RASCAL computer codes.

3.6.5 References

[3.6-1] U.S. Environmental Protection Agency, EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," Washington, DC, January 2017, https://www.epa.gov/radiation/protective-action-guides-pags.

[3.6-2] U.S. Nuclear Regulatory Commission, 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," Washington, DC, December 1978, ML051390356.

3.7 Functional Area: Materials and Reactor Component Integrity

3.7.1 Overview

The assessment of materials and reactor component integrity issues for advanced non-LWRs requires knowledge of the materials, the environmental conditions, and the loading specific to each non-LWR design. High-temperature, gas-cooled reactors (HTGR), sodium fast reactors (SFR), molten salt reactors (MSR), and other possible non-LWR designs will require unique new materials, in addition to materials commonly used in LWR designs, exposed to unique environmental and loading conditions. The one commonality is that all the assumed non-LWR technologies operate at higher temperatures than conventional LWRs.

Concerns regarding candidate structural materials for HTGRs were previously addressed using a Phenomena Identification and Ranking Table (PIRT) [3.7-1]. During past efforts to prepare for possible licensing of a HTGR, a research plan was developed [3.7-2] and implemented to address associated materials and component integrity issues [3.7-3; 3.7-4; 3.7-5]. However, much is currently unknown about the materials and structural integrity challenges for the other non-LWR designs. To address these challenges, the DOE/NE Advanced Reactors Technology Program has initiated significant research activities, and the American Society of Mechanical Engineers (ASME) has established a special division (Division 5) within Section III of the ASME Boiler and Pressure Vessel (BPV) Code Committees to address technical areas related to high-temperature reactors including material performance and qualification issues.

Thus, the primary objectives of the work in this functional area are to:

- Assess the performance needs and issues for structural materials to be used in non-LWRs, including an evaluation of gaps in data and assessment tools, through applicable domestic and international operational experience (OpE), Codes and Standards activities, and scientific and engineering knowledge.

- Support the development of a regulatory framework for materials-related issues for ANLWRs.

This effort will be facilitated, in part, through various domestic and international collaborations.

3.7.2 Work Completed in FY2017

Building on the past efforts to prepare for possible licensing of a HTGR, the staff has developed a database to aid the identification of regulatory gaps for materials and component integrity issues for SFRs and MSRs. The database includes expected environmental conditions, temperature, loadings, neutron spectrum, and expected operating life-span for various safety-significant components for each ANLWR-type. In addition, staff efforts have included literature survey and compilation of current state-of-knowledge of materials performance and component integrity for ANLWRs, gathering of information related to existing capabilities of experimental facilities and analytical tools for use in ANLWR materials testing and evaluations, and active engagement with Section III - Division 5 of ASME Codes and Standards.

3.7.3 Work in Progress

To facilitate an assessment of materials and reactor component integrity issues efficiently, the staff has awarded a commercial contract to summarize available domestic and international operational experience (OpE) for ANLWRs, including both power and research reactors. An outcome of this contract will include a technical letter report (TLR) summarizing the OpE for sodium fast reactors (SFRs) and high-temperature gas reactors (HTGRs). This TLR is expected cover the following topics for each design:

- Materials used.
- Observed and anticipated material degradation mechanisms.
- Component integrity issues.
- Possible solutions to materials and integrity challenges.
- Assessment tools (e.g., computer codes) and evaluation techniques (e.g., non-destructive examination) used to identify and address component integrity issues unique to or modified for ANLWR systems.
- Identification of specific issues based on the OpE that should be addressed within the development of regulatory infrastructure.

The NRC expects MSR technology to pose new technical challenges in the areas of materials and component integrity due to the elevated temperatures and the potentially more corrosive environment. The combination of these conditions may result in new degradation mechanisms or limit the lifetime of components. The staff has initiated a project with Oak Ridge National Laboratory (ORNL) to review existing information on candidate materials for MSRs and to identify and critique the dominant physical phenomena, materials compatibility, high-temperature corrosion under molten salt environment, and component integrity issues for MSRs. As an outcome of project with ORNL, a TLR will be developed to cover the following:

- A list and description of materials anticipated for proposed MSR designs.
- A brief rationale for use of the anticipated materials and their selection rationale (e.g., molten salt compatibility, creep resistance, high-temperature strength, irradiation testing to assess effects of neutrons and high-energy fission fragments on material degradation, etc.). This will include joints, weldments, and other fabrication-related material states.
- Expected materials and component integrity issues associated with MSR operation (e.g., molten salt corrosion, materials compatibility, etc.).
- An assessment of potential overlaps and synergies with material selection, material performance, and common degradation issues with SFRs and HTGRs.
- A preliminary list of potential gaps in material understanding for service in MSR application. (Priority ranking and confirmation will not be included as this is most appropriate following a more formal PIRT process.)
- Preliminary recommendations on potential areas that may require confirmatory research, pending formal PIRT.

3.7.4 Work Recommended for FY2018 & FY2019

The efforts related to the two projects will be completed in FY2018. The ORNL project will be part of the pre-PIRT effort for MSRs and would lead up to a more efficient PIRT-type of analysis.

A full PIRT-type analysis for materials and component integrity issues for MSRs may be essential to adequately address any regulatory gaps for this design.

The results of the ongoing efforts are intended to assist the NRC staff in prioritizing the areas requiring confirmatory research and development work to prepare for reviewing design submittals. Moreover, the identification of technical gaps will allow the NRC staff to assess potential modifications to the existing regulatory framework and to address gaps in consensus Codes and Standards relevant to licensing of ANLWRs. It is recommended that these activities continue into FY2019 to develop a prioritized plan for addressing technology gaps and supporting regulatory framework development for near-term, mid-term, and longer-term issues.

To ensure the programmatic goals are achieved efficiently, collaborative efforts will need to be developed with appropriate domestic technical experts at DOE, EPRI, and industry as well as with international technical and regulatory experts in various countries including Japan, Korea, France, Britain, and Germany through the International Atomic Energy Agency (IAEA), Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD), and other direct agreements.

3.7.5 References

[3.7-1] Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Vol. 4, “High Temperature Materials PIRTs,” 53 pp., March 2008, ADAMS accession no. ML081140462.

[3.7-2] NRC Research Plan, “High-Temperature Gas-Cooled Reactor (HTGR),” (2011), 52p. ADAMS accession no. ML110310182.

[3.7-3- Hull, A; Malik, S; Srinivasan, M; and Tregoning, R; “Survey of NRC’s Materials Research Associated with Advanced Reactors,” Proceedings from Corrosion 2012 Research Topical Symposium on Corrosion Degradation of Materials in Nuclear Power Reactors – Lessons Learned and Future Challenge, 15 pp. (2012).

[3.7-4] PNNL-20606, “Application of Acoustic Emission and Other Online Monitoring Technologies to High Temperature Gas Reactors,” 76 pp. (2012).

[3.7-5] PNNL-21180, “Literature Survey Evaluation Summary for Reliability and Integrity of Materials in High Temperature Gas-Cooled Reactors,” 162 pp. (2012).

3.8 Functional Area: Probabilistic Risk Assessment

3.8.1 Overview

The probabilistic risk assessment area consists of six contributing areas:

1. Investigate what previous work has been done in PRA for non-LWR designs.
2. Perform scoping study to understand if any work identified above can be used again or if gaps still exist.
3. Evaluate if new PRA policy will be needed to support non-LWR designs, including risk surrogates.

4. Identify any other relevant technological trends in PRA methods, models, tools, or data collection.
5. Identify guidance documents that would need to be updated to support the reviews.
6. Identify prioritized level of effort to capture any gaps identified above and prepare gap closure plans.

To approach the five contributing activities completed in FY2017, several different elements were used:

- A literature review of available risk studies, past non-LWR operating history, and other pertinent literature that could be used to identify unique design and safety issues associated with non-LWR designs and consequently could potentially impact the development and review of an HTGR PRA was conducted.

- A review of the NRC policy and guidance and standard documents to identify where existing LWR guidance and standards are applicable to non-LWR PRA as well as potential gaps was conducted.

- Methods and data used in previous non-LWR and current LWR PRAs were reviewed to assess their applicability to the non-LWR PRA and to identify where new and revised methods and data are needed.

3.8.2 Work Completed in FY2017

In FY2017, the first five contributing areas were completed. The report that documents this completion is available in the Agencywide Documents Access and Management System (ADAMS) at ML17240A228 [3.8-1]. This report explored non-LWR operational history, past non-LWR PRAs, current NRC policy and guidance for PRA and non-LWRs, the non-LWR PRA Standard, and technological trends in PRA.

The report identified potential areas that may warrant additional research, including assurance that risk-informed approaches remain consistent with NRC's integrated decision-making framework (e.g., defense-in-depth, safety margins, and monitoring) and the expanded PRA scope that is anticipated for non-LWR PRAs. Additionally, several historical policy issues were identified that may still be applicable. This information was provided to NRO to inform their path forward in Strategy 3.

3.8.3 Work in Progress

The staff is completing the prioritized gap analysis (contributing area six noted in Section 3.8.1). This will involve evaluating those areas identified in the interim report and proposing potential solutions and areas of future research that can inform the reviews of non-LWR PRAs and the NRC's integrated risk-informed decision-making framework.

3.8.4 Work Recommended for FY2018 & FY2019

In FY2018, RES proposes to complete contributing area six and prepare a gap closure plan. Any additional work in FY2018 and beyond will be closely coordinated with NRO to identify

appropriate methods to address the identified gaps. In addition, in FY2018 and FY2019, RES will continue engagement with standard development organizations to support continued development of the non-LWR standard.

3.8.5 References

[3.8-1] USNRC, "Strategy 2 of the Implementation Action Plan – Probabilistic Risk Assessment Report" (ML17240A228), September 2017.

4.0 Summary and Conclusions

The Office of Nuclear Regulatory Research began efforts to evaluate existing codes and tools and to develop codes and tools, as needed, for the regulatory review of non-LWRs in mid-2017 consistent with the NRC Vision and Strategy document and Implementation Action Plan. This included work in several areas, most notably in the areas of code selection, materials evaluation, and PRA for non-LWRs.

Work completed, or nearing completion at the end of fiscal year 2017 includes:

- An initial screening of analysis codes for design basis and beyond design basis event simulation was completed, and a suite of tools for further examination and consideration has been identified. The code suite is composed of both NRC-developed and DOE-developed codes. Future efforts will evaluate codes in the code suite against analysis requirements.

- A PIRT exercise was conducted for molten salt reactors. The PIRT focused attention on fuel salt MSRs due to their novel and unique feature of fuel being part of the coolant. The PIRT is considered preliminary in that design specifics are not available but useful in that several phenomena requiring simulation could be identified based on existing information.

- A report on Probabilistic Risk Assessment (PRA), summarizing previous work and issues for advanced non-LWRs was completed. The report identifies several policy decisions that may need to be made for non-LWRs.

Research is expected to continue in FY2018 and FY2019 with the focus on the identification of "gaps" in NRC's present capability to understand and evaluate new designs and fuels and confirm their safety under a range of operating conditions.