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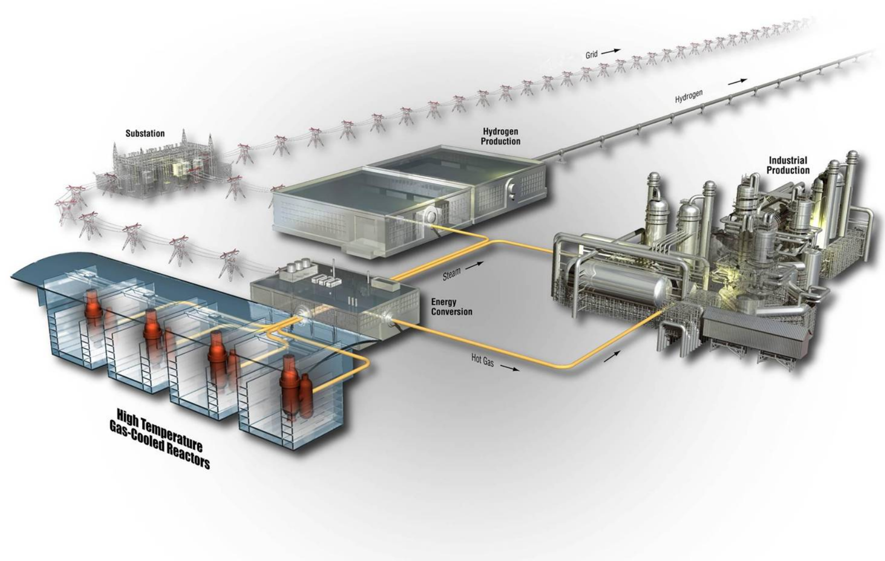
Advanced Reactor Technologies - Regulatory Technology Development Plan (RTDP)



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REVISION LOG

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0	05/18/2015	All	New issue.
1	09/08/2017	All	General technical update of entire document

SUMMARY

All U.S. commercial nuclear reactor designs undergo a comprehensive safety assessment conducted by reactor technology developers and the U.S. Nuclear Regulatory Commission (NRC). As the independent regulatory agency responsible for commercial nuclear plant licensing, the NRC also conducts key confirmatory research on nuclear safety. However, the primary focus of an NRC licensing review is associated with evaluating information submitted to the agency in a license application.

The U.S. Department of Energy (DOE) is a government agency that assists suppliers in performing essential research and development (R&D) for new reactor technology. A wide variety of tests, studies, and investigations are sponsored by DOE that address important system performance parameters and the validation of analysis methods and tools needed to perform safety reviews.

Data and information produced through DOE-sponsored R&D are often crucial components in successful licensing. Consequently, the test plans and conclusions generated by DOE-sponsored research must consider regulatory requirements as tests are being planned and performed. Well-informed R&D planning helps ensure DOE-sponsored research adequately addresses licensing challenges arising later during the application process.

The Advanced Reactor Technologies (ART) Regulatory Technology Development Plan (RTDP) links major research activities in advanced non-light water reactor technologies (as sponsored by the DOE Office of Nuclear Energy, Science, and Technology’s [DOE-NE] ART program) to key regulatory requirements and licensing concerns likely to impact advanced reactors. As a consequence of current ART priorities, the RTDP is focused on three different types of advanced reactors - the modular high-temperature gas cooled reactor (HTGR), the sodium-cooled fast reactor (SFR), and the molten salt reactor (MSR) concepts.

Linking current reactor technology R&D to licensing requirements early in R&D planning is a complex undertaking that requires interaction and coordination with reactor suppliers, NRC staff, and academic- and government-sponsored researchers working to bring concepts to maturity. The RTDP was created in 2015 to aid that linkage and further NRC’s Advanced Reactor Policy Statement of 2008 that was again restated in NRC’s 2012 Report to Congress on Advanced Reactor Licensing. This statement strongly encouraged reactor researchers and developers to seek out new and improved safety and security features and called for proposals that are simplified, inherent, and passive as a means to accomplish essential safety and security functions. Furthermore, it is expected that such information would be presented to NRC staff for review and feedback during the pre-licensing phase of application development to help assure confirmatory testing is done adequately, to provide for collection of data sufficient to validate computer codes and analysis methods, and show that system interaction effects are acceptable.

Section 3 of this document identifies discrete DOE/ART R&D activities underway, planned, or potentially necessary to license modular HTGR, SFR, and MSR technologies. Topics of concern are identified and discussed based on ART research plans, ART program leadership opinions in technical R&D areas, pre-licensing precedents derived from recent feedback from NRC staff, and interactions with the advanced reactor community. Insights on the regulatory implications associated with certain research and effects on the “critical path” licensing timeline are also provided. These activities are prioritized with respect to greatest regulatory need and/or technology safety case development; Activities with very long lead-times or significant sequential dependencies with other work are noted and ranked accordingly.

Section 4 identifies eight licensing priority recommendations for ART planning consideration. These recommendations, established using information listed in Section 3 tables, consist of:

Recommendation 1: Continue recovery, archiving, and configuration control of SFR information from the Experimental Breeder Reactor-II (EBR-II) and the Fast Flux Test Facility (FFTF); preliminarily qualify recovered information according to Nuclear Quality Assurance (NQA-1) requirements.

Recommendation 2: Identify gaps in SFR metallic fuel knowledge and plan tests to close critical gaps and reduce uncertainty.

Recommendation 3: Complete the Advanced Gas Reactor (AGR) Fuel Test Plan and the Graphite Technology Development Plan.

Recommendation 4: Establish the role of MSR fuel in plant safety; develop a definition of fuel qualification appropriate for mechanistic source term (MST) development in MSRs.

Recommendation 5: Continue development, qualification, and validation of safety analysis codes and methods compatible with modular HTGR and metallic fuel SFR designs.

Recommendation 6: Complete experimental tests at the High Temperature Test Facility (HTTF) and the Natural Convection Shutdown Heat Removal Test Facility (NSTF).

Recommendation 7: Create “generic” MSR R&D activity sets for a “standard” design to increase licensing readiness for the entire MSR technology class.

Recommendation 8: Establish fundamental cross-cutting instrumentation and control (I&C) system requirements for advanced reactors and develop a plan to address new performance and reliability requirements.

Section 5 identifies additional topics expected to emerge as a future licensing priority but are not yet on or near the critical path for deployment. Adequately resolving these issues may eventually require ART support, but for now, those R&D activities are not yet barriers in license application development.

The reader is advised that the RTDP is not a “roadmap” in advanced reactor licensing, nor is it meant to replace a design-specific licensing plan. Instead, this document focuses on evaluating ART R&D activities and opportunities and communicating the significance of that work in terms of importance to licensing. Its primary purpose is to inform R&D planners by identifying the needs of both prospective license applicants and the NRC safety reviewer.

Applicants are responsible for writing a licensing plan tailored to the design details and safety approach characteristics of that proprietary design. The RTDP is a tool available to ART program managers and principal investigators to coordinate and interface their R&D work with regulatory requirements until such time that a design-specific licensing plan is written and available.

This RTDP revision updates prior information related to modular HTGR and SFR technology and adds a regulatory effects analysis for MSR technology R&D. The RTDP will be further adjusted as needed to meet the needs of ART research planning.

CONTENTS

FIGURES

TABLES

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADTR	Advanced Demonstration and Test Reactor
AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ARC	Argonne Computation Code
ART	Advanced Reactor Technologies (program)
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
BDBE	beyond design basis event
BPV	Boiler and Pressure Vessel
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
COL	combined license
CP	construction permit
CRBR	Clinch River Breeder Reactor
CSDRS	certified seismic design response spectrum
DBA	design basis accident
DBE	design basis event
DC	design certification
DID	defense in depth
DOE	U.S. Department of Energy
DOE-NE	DOE Office of Nuclear Energy, Science, and Technology
dpa	displacement-per-atom (rate)
DRACS	Direct Reactor Auxiliary Coolant System
DTF	designed-to-fail
EBR-II	Experimental Breeder Reactor-II
EPA	Environmental Protection Agency
ESP	early site permit

FCT	Fuel Cycle Technologies (program)
FFTF	Fast Flux Test Facility
FHR	high-temperature fluoride salt reactor
FIRS	foundation input response spectrum
FPT	fission product transport
FQ	fuel qualification
GA	General Atomics
GAIN	Gateway for Accelerated Innovation in Nuclear
GFR	gas-cooled fast reactor
HMI	human-machine interface
HTGR	high-temperature gas cooled reactor
HTTF	High Temperature Test Facility
HTTR	High-Temperature Test Reactor
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
iPWR	integral pressurized water reactor
IRP	Integrated Research Projects
JAEA	Japan Atomic Energy Agency
LBE	licensing basis event
LFR	lead-cooled fast reactor
LMP	Licensing Modernization Project
LMR	liquid metal-cooled fast reactor
LOCA	loss-of-coolant accident
LWA	limited work authorization
LWR	light water reactor
M&S	modeling and simulation
M&TE	measuring and test equipment
MC&A	material control and accountability
METL	Mechanisms Engineering Test Loop
MHTGR	Modular High Temperature Gas-Cooled Reactor
MSR	molten salt reactor

MSR-E	Molten Salt Reactor Experiment
MST	mechanistic source terms
MW(t)	megawatt (thermal)
NEAMS	Nuclear Energy Advanced Methods and Simulation
NEET	Nuclear Energy Enabling Technologies (program)
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Plant
NLSSI	non-linear soil-structure interaction
NRC	U.S. Nuclear Regulatory Commission
NSTF	Natural Convection Shutdown Heat Removal Test Facility
NQA	Nuclear Quality Assurance
QAPD	quality assurance program description
QAPP	quality assurance program plan
OECD	Organization for Economic Co-operation and Development
OL	operating license
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PARCS	Purdue Advanced Reactor Core Simulator
PIE	post-irradiation examination
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwestern National Laboratory
PRA	probabilistic risk assessment
PRISM	power reactor innovative small module
QA	quality assurance
QAPD	quality assurance program description
R&D	research and development
RCCS	Reactor Cavity Cooling System
RG	regulatory guide
RPV	reactor pressure vessel
RTDP	regulatory technology development plan
RVACS	reactor vessel auxiliary cooling system
SARRDL	Specified Acceptable Core Radiological Release Design Limit
SCALE	Standardized Computer Analyses for Licensing Evaluation

SDO	standards development organization
SER	Safety Evaluation Report
SFR	sodium-cooled fast reactor
SI	seismic isolation
SNL	Sandia National Laboratories
SNM	special nuclear material
SPRA	seismic probabilistic risk assessment
SQA	software quality assurance
SSC	structure, system, and component
SSI	soil-structure interaction
TDO	Technology Development Office
TI-RIPB	technology-inclusive, risk-informed, and performance-based
TREAT	Transient Reactor Test Facility
TRISO	tristructural isotropic
TRL	technology readiness level
V&V	verification and validation
VHTR	very high temperature reactor

Advanced Reactor Technologies - Regulatory Technology Development Plan (RTDP)

1. INTRODUCTION

The Advanced Reactor Technologies (ART) Regulatory Technology Development Plan (RTDP) links advanced non-light water nuclear reactor (non-LWR) technology development activities sponsored by the U.S. Department of Energy (DOE)'s Office of Nuclear Energy, Science, and Technology (DOE-NE) ART Program to regulatory requirements and key licensing issues likely to impact entries into the domestic commercial energy market. The discussions and recommendations contained in the RTDP are not constrained to a particular category, class, or type of non-LWR, but can be applied to arrays of licensing-related concerns as dictated by current ART research and development (R&D) priorities and technology enhancement opportunities. However, because ART is now primarily focused on research that brings three types of non-LWR reactor designs to deployable maturity, the RTDP is also scoped to reflect that emphasis.

Commercial nuclear power reactors in the U.S. are licensed after successfully completing an independent safety assessment conducted by the U.S. Nuclear Regulatory Commission (NRC). This assessment must result in findings that the information contained in the plants' license application is comprehensive, representative, and adequately characterizes systems and operations that are protective of public safety. As a regulatory agency, the NRC does not conduct technology development research on new reactor designs, but rather focuses on evaluating and confirming information and safety conclusions submitted by applicants to secure a construction permit (CP), operating license (OL), early site permit (ESP), limited work authorization (LWA), design certification (DC), and/or a combined license (COL).

Information essential to a complete license application often comes from sources other than the applicant. As a governmental agency tasked with performing R&D that facilitates new reactor technology, DOE-NE sponsors a wide range of studies and investigations essential to licensing success. Some of this information is foundational to reactor and balance-of-plant system performance, safety, and structure, system, and component (SSC) reliability. Accordingly, considerable R&D done by DOE-NE will be compared to regulatory technical requirements during future licensing actions, thereby making those requirements an important consideration during planned research and performance.

The NRC has developed a large body of technical requirements and guidance based on large LWR power plant experience. However, many elements of those requirements cannot be easily or clearly translated into non-LWR applications. To aid non-LWR suppliers in licensing, NRC, DOE, and industry are engaged in a regulatory framework modernization effort. When completed, the results of this effort may substantively affect the way technology-enabling R&D is planned and performed. Additional information on this topic, along with current status, can be found in numerous NRC and industry stakeholder position papers posted on the NRC website.

It is also important to note that all R&D used to support assessments of safety must be done in accordance with NRC-endorsed quality assurance (QA) requirements. Adequate confidence in test results is critical to licensing decisions and therefore requires that proper quality controls be implemented as research plans are being written. In general, this means DOE-NE-sponsored technology development plans should implement quality assurance and administrative control requirements that meet Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The NRC also allows use of standards described in Nuclear

Quality Assurance (NQA)-1-2008/1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications," as endorsed through Regulatory Guide (RG) 1.28, Revision 4, "Quality Assurance Program Criteria (Design and Construction)," to meet this requirement. Establishing and implementing appropriate and endorsed QA measures early in experimental and testing protocols is critical to assure data generated by R&D tests important to safety can later be used in licensing decisions.

1.1 Purpose

The ART RTDP seeks to identify critical regulatory issues pertaining to ART R&D activities and direct attention to planning needed to address licensing. Since licensing-significant research sponsored by DOE-NE is conducted at various DOE national laboratories, universities, and other research organizations, these entities must be adequately informed that the research being conducted may be required to meet and document compliance with formally established standards of accuracy, thoroughness and quality appropriate to nuclear safety studies. It should be remembered, however, that not all reactor technology development R&D carry a significant licensing implication; non-safety related R&D are not emphasized in the regulatory effects analysis of the RTDP.

Figure 1 illustrates how the RTDP targets overlapping interests between ART R&D and the nuclear regulatory environment.

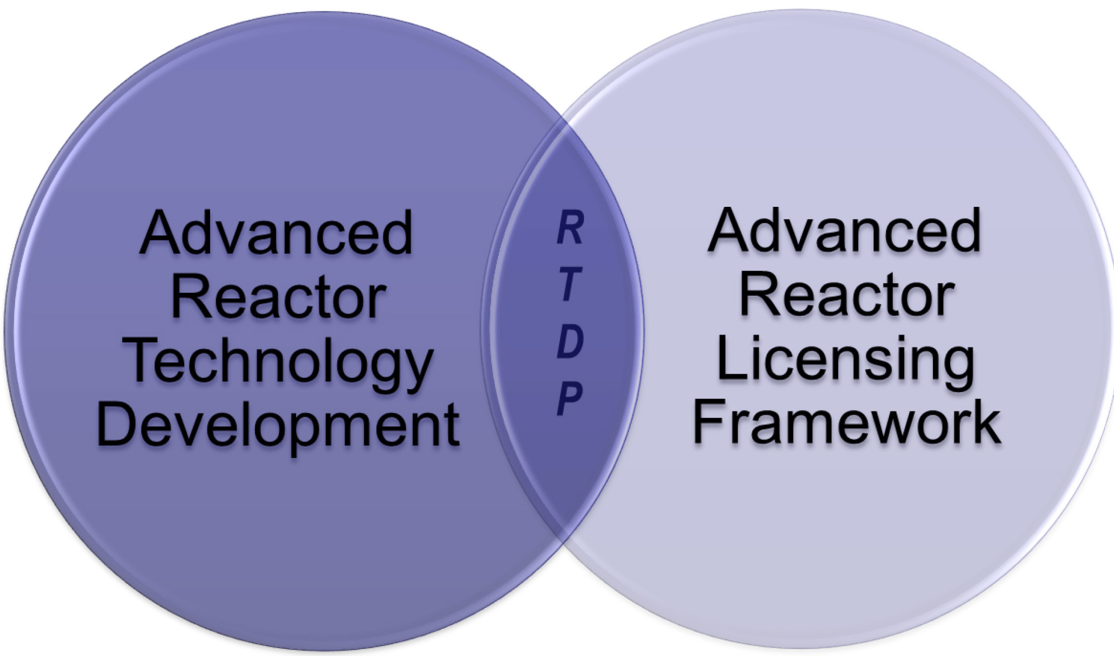

Figure 1. RTDP linkage between advanced reactor R&D to licensing.

Early interactions between technology developers, NRC staff, and (sometimes) ART researchers may be needed to identify, clarify, and properly apply regulatory requirements to technology development plans. Consideration of licensing needs during initial R&D planning has been recognized by the NRC as an important technology development program concern, especially for radical new reactor concepts. As discussed in the 2008 NRC "Advanced Reactor Policy Statement" (73 FR 60615) and restated in NRC's "2012 Report to Congress on Advanced Reactor Licensing,"[1] advanced reactor research should be planned to include tests of new safety or security features that differ from existing operating reactor designs and/or use new simplified, inherent, and/or passive means to accomplish safety or security functions. Appropriate testing should be conducted to demonstrate new features perform as predicted, provide for collection of sufficient data to validate computer analysis codes, and show system interaction effects are acceptable.

The NRC policy statement strongly encourages design innovations that enhance safety, reliability, and security. However, technology developers seeking to employ such innovations must prove them to be safe, reliable, and secure by means of a straightforward development program. The statement further notes that absent significant operational experience for a new feature, a plan to innovatively deploy a demonstration-level reactor and/or establish new technology development programs should be presented to the NRC for review as early as possible. Early interaction allows NRC staff to assess and advise how a proposed program should be implemented to satisfy regulatory requirements.

It is expected that design-specific information essential to independent safety reviews will be unavailable during early stages of development. Often, preliminary presumptions about a safety basis must be made to facilitate test planning in areas such as fuel qualification (FQ), mechanistic source term (MST) development, and the qualification of new materials in new systems and new applications. Accordingly, the RTDP seeks to help identify, assess, and recommend priorities for ART R&D activities and opportunities based on associated regulatory effect. It does this with consideration for minimizing the critical path timeline for licensing. A corollary objective is to assure research is coordinated with the safety assessment needs of applicants and NRC safety reviewers.

1.2 Advanced Reactor Technologies

DOE-NE has published a vision and strategy for making advanced reactors a major energy resource.[2] This document projects that by the year 2050, advanced reactors will provide a significant and growing component of the nuclear energy mix both domestically and globally. This will happen as a consequence of their advantages in improved safety, cost, performance, sustainability, and reduced proliferation risk. To support this vision, a goal was established that by the early 2030s at least two non-light-water advanced reactor concepts will reach technical maturity and demonstrate safety and economic benefits through actual operations. It will do this by successfully completing a licensing review by the NRC sufficient to allow subsequent construction and operation of additional commercial units.

Six types of advanced reactors are identified in the DOE vision. These are modular high-temperature gas-cooled reactors (HTGRs), gas-cooled fast reactors (GFRs), sodium-cooled fast reactors (SFRs), lead-cooled fast reactors (LFRs), molten-salt reactors (MSRs), and high-temperature fluoride salt reactors (FHRs). Depending on the technology maturity of each concept, early 2030s demonstration plan targets can be divided into *commercial* demonstration and *engineering* demonstration. For advanced reactors already demonstrated on an engineering or proto-commercial scale, the target is commercial demonstration. For concepts that have not demonstrated power production on an engineering scale, the target is engineering demonstration. Of the above technologies, only SFRs and modular HTGRs are generally considered viable commercial demonstrators by the early 2030s while remaining concepts are likely engineering demonstrators.

DOE-NE is actively supporting three advanced reactor classes at this time. Support is typically in the form of technology R&D testing and covers HTGR, SFR, and MSR design approaches. As a consequence of this emphasis, the RTDP is focused on analyzing the regulatory affects associated with the technology R&D. The following subsections summarize key technology elements and attributes.

1.2.1 High Temperature Gas-Cooled Reactors

The graphite-moderated modular HTGR possesses a relatively high degree of technical maturity for a non-LWR. Extensive in-pile testing and engineering demonstration experience dating back to the 1960s are available along with commercial demonstrations in the 1980s. Recent industry interest is focused on prismatic block and pebble bed core concepts with a lower (<750°C) reactor outlet temperatures. While a HTGR design could be commercially operational within 15 years, additional R&D is needed to support long-term very high temperature reactor (VHTR) operations with higher outlet temperatures (<950°C).[3]

HTGR safety is founded on tristructural isotropic (TRISO) fuel particles bonded in a graphite matrix to form either a cylindrical compact or a spherical pebble (see Figure 2). TRISO-coated fuel particles consist of multiple layers that act in series to provide a miniature containment structure limiting radioactive fission product release. The fuel design contains a fuel kernel surrounded by porous carbon, inner and outer pyrolitic carbon layers, and a silicon carbide layer. A buffer layer allows limited kernel migration and provides some retention of gas compounds. The silicon carbide layer ensures particle structural integrity and helps retain metallic fission products. Compacts are inserted into hexagonal graphite blocks to assemble a prismatic fuel element while a 5-mm layer of graphitic matrix material forms a protective shell around the inner fueled zone of pebbles.

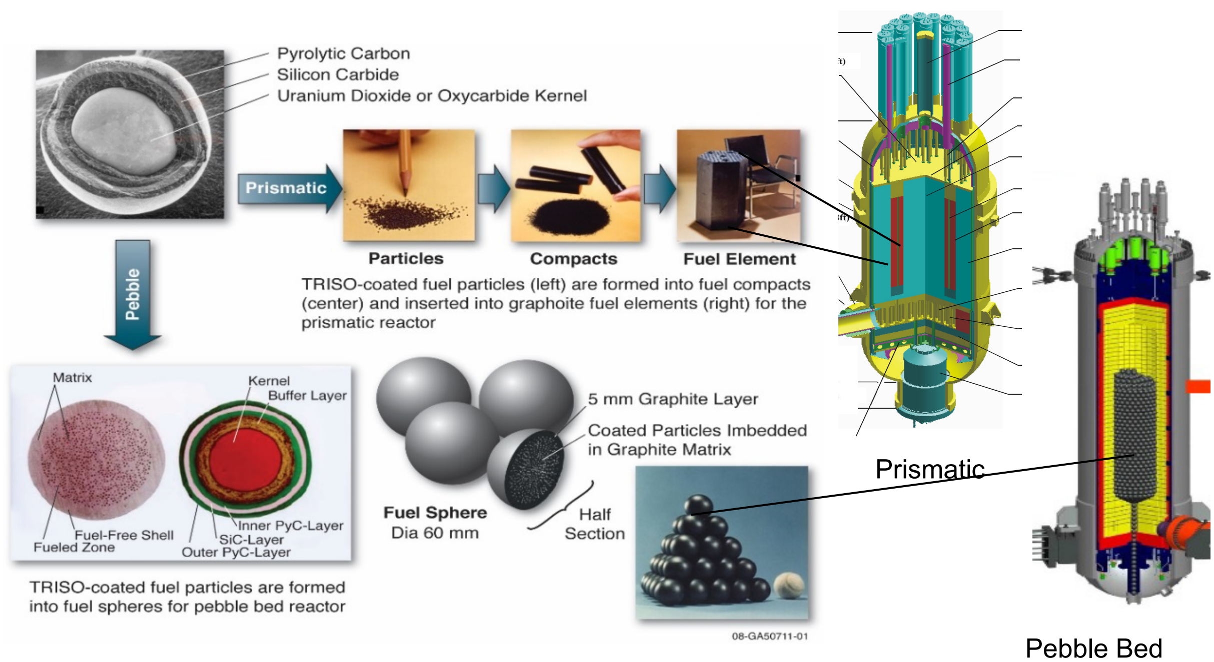

Figure 2. TRISO fuel design.

HTGRs have been a commercial power venture from the beginning and subject to decades of R&D. The Peach Bottom Atomic Power Station (115 thermal megawatt [MWt]) was ordered by the Philadelphia Electric Company from General Atomics (GA) and operated as a prototype from 1966 to 1974. Fort St. Vrain (also a GA design) used an early version of TRISO fuel (highly enriched uranium and thorium) to demonstrate the technology. In the 1990s, GA designed the 350 MWt Modular High Temperature Gas-Cooled Reactor (MHTGR) and received a pre-application safety evaluation from the NRC.[4] More recently, the NRC reviewed key modular licensing issues as part of DOE's Next Generation

Nuclear Plant (NGNP) project.[5] As a consequence of these earlier efforts, considerable licensing history is available and high degrees of confidence accompany remaining modular HTGR R&D needs.

1.2.2 Liquid Metal-Cooled Fast Reactor

Fast neutron reactor coolants have been studied for over seven decades. Most recently, DOE reviewed the state of overall fast reactor design and different liquid metal coolants in a "roadmap" outlining technology readiness and developmental paths for two types of liquid metal-cooled fast reactor (LMR) technologies, i.e., SFR and LFR.[6] Of the two, SFR is the most technologically mature and nearest commercial licensing readiness.

Both metal and oxide fuels have been used in SFRs and can support commercial demonstration planning. The first power-generating SFR in the U.S. was the Experimental Breeder Reactor-II (EBR-II), operated from 1963 to 1994. It was designed to produce 62.5 MWt using metal fuel. The facility was permanently decommissioned in 1998. The initial EBR-II mission was to demonstrate the fuel breeding capabilities of a fast reactor, but expanded to the testing of fuels and materials, as well as to demonstrate a closed fuel cycle and inherent safety features during reactor transients. The inherent safety of EBR-II was demonstrated in 1986 through a series of unprotected transient experiments, which included disconnect of electrical supplies to primary reactor coolant pumps without reactor scram. This disabled emergency shutdown of systems and primary coolant pumps. The subsequent temperature increase led to core expansion and sub-criticality via neutron leakage. Decay heat was removed by natural heat transfer mechanisms through a Direct Reactor Auxiliary Coolant System (DRACS) and the plant safely shut itself down.

EBR-II was an engineering demonstration reactor built and operated by DOE. The Fast Flux Test Facility (FFTF) reactor (also a DOE plant) used oxide and metal fuel. FFTF, along with Fermi-I (a metal fuel design), were performance demonstration reactors. FFTF also served as a material test reactor without an energy conversion system. Relatedly, the Clinch River Breeder Reactor (CRBR) and power reactor innovative small module (PRISM)/Mod-A would have been commercial demonstrations had they been built. Despite extensive interactions with NRC on these and other SFR designs in the 1970s, 1980s, and 1990s, SFRs received only limited regulatory review and none led to NRC license issuance.[a]

Fast reactors using metallic fuel immersed in a sodium pool are actively undergoing technology support development within DOE, industry, and universities. Key objectives in modern SFR design includes enhanced reactor performance using very-high-burnup fuels, advanced cladding and structural materials, and cost-saving methods, such as compact power conversion systems. Examples of emerging SFR designs that could be licensed by the 2030s are PRISM, the 250-MWt Advanced Reactor Concept (ARC-100), the prototype Traveling Wave Reactor (TWR-P), and the 1000-MWt Advanced Burner Reactor (ABR).

1.2.3 Molten Salt Reactor

Molten salt depicts another class of Generation IV fission reactors where the primary coolant, or the fuel itself, is a molten salt mixture. MSRs generally run at higher temperatures than water-cooled reactors and produce higher thermodynamic efficiencies while remaining at a low vapor pressure. Advantages of MSR technology include high power density, low operating pressure, low stored energy, a prompt negative temperature coefficient, and capabilities for continuous fueling and fission product removal.

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a. Fermi-1 was licensed by the Atomic Energy Commission rather than the NRC. While NRC did review FFTF to validate a safety review process, it did not receive a NRC license.

Disadvantages include generation of fission products/transmutation products and high temperature challenges to materials.[7]

There is a large spectrum of MSR design variations. Major differences include a fast vs thermal neutron spectrum, breeder vs burner, liquid vs solid fuel, thorium vs uranium fissile material, and coolant choices ranging from salt to gas and metals. The most common MSRs discussed today fall into three general classes:

1. Solid-fuel where the fuel is cooled by a separate, non-fueled primary circuit salt
2. Salt-fuel where fuel is dispersed in salt flowing through the primary system using fluoride salts
3. Salt-fuel where fuel is dispersed in salt flowing through the primary system using chloride salts.

Salt-cooled and salt-fueled concepts exist with fast, epithermal, or thermal neutron spectrums. Salt-cooled designs using fluoride compounds consider mainly thermal spectrums and are denoted as FHRs, which are the subject of considerable technology development interest and may be able to utilize ART R&D on TRISO fuels and fixed-core moderating graphite. Fast spectrum salt-fueled concepts may use either fluoride or chloride salts and generally do not require in-core moderating materials.

Most MSR technical experience is derived from the Molten Salt Reactor Experiment (MSR-E) that operated from 1965-1969 at Oak Ridge National Laboratory (ORNL). This 7.4 MW(t) engineering-scale test reactor was cooled with molten fluoride salt and used ^{235}U and (later) ^{233}U fuels. The test program generated a wealth of technology development information for that design variant. Fast spectrum salt-fueled concepts, especially those that rely on chloride salts, are much less developed with relatively little irradiation performance data currently available.

1.3 R&D Applications

DOE-sponsored R&D is an essential resource for bringing non-LWR concepts to technological maturation. This resource is also essential to assessing concept safety and successfully licensing the technology for commercial use. With respect to licensing, new reactor technology development studies must be performed not only to generate characterization information and validate the reactor design safety basis but also addresses the needs of the independent safety review process as performed by NRC staff.

The independent safety review enables a decision to certify a new reactor design as acceptably safe and issue a license to build and operate the nuclear plant. These decisions will be guided by scientific and engineering study findings that confidently demonstrate risks to public health are acceptable and that overall safety and common security are not threatened. Assessments that support such a finding will be done based on comprehensive technical evaluations and consequence predictions covering design basis events, the safety SSCs employed, methods of proposed operation, accident prevention and consequence mitigation, and barriers to limit radioactive material release. Calculated radiological dose to offsite receptors as a consequence of postulated and bounding event releases is emphasized during a regulatory safety assessment and is the ultimate criteria by which decisions are made to grant a license.

The technical criteria available to evaluate plant design and operations safety was developed over 50 years ago and validated mostly with research and experience pertaining to large LWRs generating baseload electric power for the grid. These criteria may be of limited use or not applicable to non-LWR designs. Furthermore, LWR-oriented requirements and analysis tools that ascertain compliance with regulatory requirements may not be easily translated for applications in some advanced reactor designs, as was the case with the NGNP.[8] Efforts are currently underway to modernize key elements of the current advanced reactor licensing framework by making it more technology-inclusive, risk-informed, and performance-based (TI-RIPB).[9,10]

2. SCOPE OF TECHNOLOGY RESEARCH

The RTDP identifies, evaluates, and prioritizes areas of R&D concern relative to future licensing by individual advanced reactor technology categories. A regulatory effects analysis system has been devised whereby high-level licensing requirements are overlaid on ART R&D activity descriptions to enable comparison and semi-quantitative assessment. Doing this allows research planners to be informed of licensing perspectives related to the R&D activity and aids in communicating the implications concerning NRC safety reviews. It is important to note that recommended licensing priorities derived from the RTDP regulatory effects analysis consider not only the current state of topical knowledge related to that R&D topic, but also the anticipated timeline for completing the licensing critical path.

As R&D activities important to plant safety are planned and performed, it will be necessary to engage in pre-licensing dialogs with stakeholders. These interactions typically start by soliciting inputs and priorities from technology developers, reactor designers, vendors, NRC staff, codes and standards development organizations, and similar entities through workshops focused on R&D requirements and approaches that resolve experimental designs, communicate test outcomes, and derive final conclusions.

Eventually, applicants must develop a licensing plan for their particular design. This plan solidifies design-specific compliance strategies and identifies R&D still needed to address requirements. The plan couples specific regulatory criteria to the plant safety basis and ensures appropriate safety assessment methods and tools are available to demonstrate compliance. The RTDP serves as the ART R&D “Licensing Plan” and aids technology development planners in understanding the impact of their investigation on licensing as well as identify necessary interfaces until such time as a licensing plan specific to a proprietary design becomes available.

2.1 Key Research Areas

Many kinds of R&D are needed to establish and confirm new reactor safety. Availability of verified and validated analytical safety tools becomes a very significant licensing obstacle if not addressed during technology R&D planning. If appropriate analysis methods are unavailable or if their validity cannot be confirmed to a degree that supports a safety conclusion, a license application will not be accepted by NRC for review (let alone be granted a license).

The NRC will perform an independent regulatory safety assessment for all normal and off-normal design plant conditions. The analysis relies heavily on thermal-fluid and neutronic (reactor physics) attributes of the technology. Major analysis topics include:

- Accident progression modeling
- Primary system and containment performance
- Fission product behavior modeling
- Core heat removal
- Thermal-fluid dynamics
- Nuclear analysis
- Fission product transport

- Initiating event frequency.

Every reactor technology type is required to have appropriate methodologies, analytical tools, and high-quality data available to answer questions and scenarios related to plant safety. These challenges can be organized according to three basic functions:

1. Adequate core heat removal—

   Challenges to heat removal involve timely and sufficient cooling of fuel elements, the core, the reactor vessel, and other design elements critical for radionuclide retention. Assuring fission product barrier integrity is a crucial safety priority and will be closely scrutinized during NRC's review. Back-up systems may be needed to assure adequate defense in depth (DID) for required safety functions during anticipated operational occurrences and design basis accident events.

2. Reactivity control—

   Challenges to reactivity control involve maintaining the reactor in a stable condition. A design may employ passive physics (e.g., negative temperature coefficient) to back up active control elements designed to handle challenges to reactivity control. It must be demonstrated that reactivity control features will perform as intended in all circumstances as needed to maintain safety.

3. Control of radionuclide release—

   Challenges to systems that assure retention of radionuclides involve maintaining fuel integrity, upholding core structures, and strengthening the integrity of barriers that limit release of radioactivity to the environment. In a most basic sense, the NRC safety review is focused on assuring the public is protected from risks associated with offsite radionuclide releases from the plant.

It should be noted that advanced reactor suppliers that employ highly innovative fuel designs (e.g., thorium instead of uranium) and/or new methods to assure reactor core cooling (e.g., a molten salt as a heat transfer fluid) in combination with other new active or passive safety features must still fully characterize and evaluate elements of thermal-fluids behavior, neutronics, and fission product behavior. Prototype demonstrations of a design may be mandated to develop integrated systems and analysis tools with the requisite confidence and fidelity to satisfy safety reviewers.

Major R&D areas relative to a plant safety review and licensing is summarized in Figure 3.

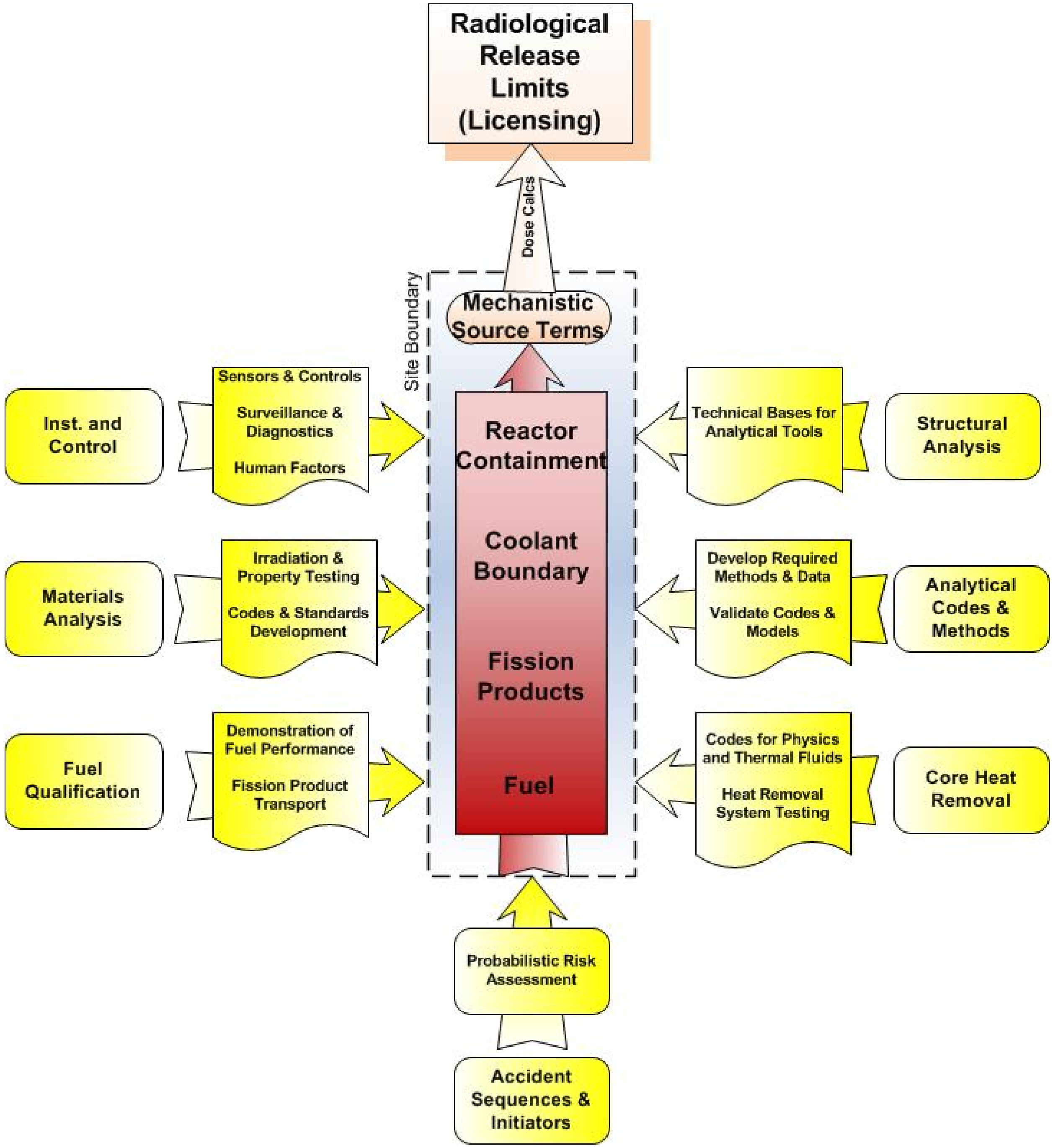

Figure 3. R&D elements important in plant safety and licensing reviews.

Inspection of Figure 3 reveals the ultimate licensing objective is meeting offsite radiological release limits. Deterministic approaches now used by LWR developers for radionuclide release analysis are generally not well suited or applicable to a non-LWR design. Realistic radionuclide release analyses (with adequate margin) of all factors influencing offsite dose calculations are now expected by regulators of advanced reactor applicants. This analysis must be based on objective test information concerning fuel behavior during all plant design conditions, fission product release, hold-up and transport characteristics during bounding design conditions, and be predictively modeled to show attainment of offsite radiological release limits.

The following subsections further elaborate the role R&D plays in addressing licensing issues.

2.1.1 Fuel Qualification

The methods in nuclear fuel design, manufacture, and use are foundational to plant safety assessments. Extensive testing and characterization information is needed to show a fuel performs in a manner that assures applicable regulatory criteria are met. Performance demonstration and qualification of new reactor fuels typically require very long duration R&D, the challenge of which is amplified by the need for sophisticated and specialized infrastructure to collect data. Often these capabilities are relatively scarce, insufficient, or otherwise inaccessible. As a consequence of nuclear fuel testing infrastructure problems, DOE facilities have become a leading resource for planning and performance of reactor fuels-related research.

Formal fuel qualification programs encompassing short- and long-term irradiations for new or modified fuel forms are often conducted at DOE facilities. Tests must be done under stringent QA protocols recognized and accepted by the NRC. The complexity associated with establishing an acceptable QA program, coupled with the long lead-times needed to perform irradiations and post-irradiation examinations (PIE), typically cause ART fuels-related research and qualification to be a significant and ongoing licensing concern.

A robust experimental database is needed to allow technology developers and NRC staff to understand fuel system design characteristics and responses to a full range of fuel burnup conditions. This understanding must also be able to support accurate simulations of fuel performance and fission product transport (FPT), retention, and release estimation to the environment under the full spectrum of normal and off-normal conditions, including accident scenarios.

A regulatory effects analysis of ART-related research regarding advanced reactor fuel testing and qualification is provided in Table 1. Licensing-oriented priority recommendations and observations regarding advanced reactor fuel development and qualification are provided in Subsection 4.1 and in Section 5.

2.1.2 Mechanistic Source Terms

A "source term" refers to normal and off-normal releases of radionuclides that originate from the fuel and are transported throughout the plant to the off-site environment. With respect to advanced reactor technologies, NRC advocates a "mechanistic source terms" (MST) approach to radionuclide release estimation. The MST approach focuses on realistically modeling actual and postulated releases and transport of radionuclides from the source to potential receptors for specific plant licensing basis event (LBE) scenarios. The model must account for retention and/or transmutation phenomena and consider uncertainties and unknowns associated with the process. Determining a MST for a new reactor design involves complex phenomena modeling that must be characterized on the basis of extensive test data and well-developed simulations for all mechanisms of significance. While development of a technically sound MST is somewhat design-specific and ultimately the responsibility of the applicant, analysis approaches, tools, and methods used in MST-related assessments may be more generic and therefore of use to an entire class of design.

Radionuclide releases must be defined starting at the source (i.e., the fuel) and quantified with respect to transport behaviors and attenuation factors of release paths to the environment. Knowing fission product retention characteristics behind barriers (as a function of time) is critical to MST assessments. Generation and release of key fission products during LBEs will be addressed in part through irradiated fuel tests. These and other tests will be planned and performed with an underlying goal of quantitatively understanding the phenomenology regarding fission product release and enable consequence prediction at established points of compliance.

R&D related to MST development is closely related to and highly reliant upon activities that characterize fuel performance. This is because reactor fuel is the "starting point" for fission products that must undergo MST analysis. The licensing effects analysis conducted relative to ART research for MST development is provided in Table 1 (in conjunction with fuels information). Licensing-oriented recommendations and observations affecting MST research are discussed in Subsections 4.1, 4.2 and Section 5.

2.1.3 Analytical Codes and Methods

Analytical code development and verification and validation (V&V) activities are essential to safety evaluation processes collectively known as "assessment." Developing assessment methods to address reactor safety is usually resource- and time-intensive, but critical to licensing success. Budget realities and deployment timelines now being proposed make it generally not viable for any single organization to undertake development of all required assessment tools for a given reactor class. Therefore, early collaboration and cooperation between advanced reactor suppliers, DOE, NRC staff, and other stakeholders is essential to develop robust tools that can commonly serve related non-LWR types.[11]

While some assessment tools can be used for "trend confirmation," detailed assessment tools optimized to address key safety-significant elements for each design class must also be available. The assessment of safety-significant phenomena and events must also have an appropriate level of fidelity, resolution, and conservatism that support NRC standards. Often, key phenomena important to safety are initially identified through expert panel elicitations. However, once candidate parameters are identified as a potential concern, data must be generated that can be used to V&V tools and assure assessments do not exceed acceptable levels of intrinsic uncertainty.

NRC has analysis codes for conventional and advanced LWRs that could be adapted for use in non-LWR applications. For non-LWR reactors, however, initial development tasks must include an evaluation and down-select of these or other codes for regulatory use. This is especially true for designs with little regulatory assessment history and only limited prior code development effort. The staff's current emphasis on advance reactor analytical assessment tool development is to leverage (to the maximum extent practical) collaboration and cooperation with domestic and international stakeholders to establish sets of tools and data that can be shared, understood, and accepted by NRC, technology vendors, and international regulatory partners. Having a common understanding of highly qualified tools and support data (as opposed to developing that understanding for each license application review), significantly improves efficiencies and reduces cost for all involved.

Common assessment tool development efforts could start with confirmatory calculations of reactor kinetics and criticality (often conducted by the staff) using the Purdue Advanced Reactor Core Simulator (PARCS) and Standardized Computer Analyses for Licensing Evaluation (SCALE) codes. Confirmatory analyses of HTGR assessment approaches were done during NGNP and resulted in extensive code development work that others in the non-LWR technical community may choose to adapt for use in reactor kinetics and criticality analysis. While PARCS has also been developed to support SFR design analysis by NRC's international regulatory partners, it is uncertain if that will be adopted to support analysis for other designs.

Using the aforementioned code examples, confirmatory analysis capabilities pertaining to reactor kinetics and criticality safety, estimates can be further refined with respect to regulatory assessment needs. A comprehensive functional need assessment for a non-LWR modeling application could include:

- Determination of conditions and transients to be modeled
- Determination of important phenomena that must be modeled through development of a Phenomena Identification and Ranking Table (PIRT)
- Assessment of existing reactor core analysis and criticality safety capabilities in 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 computer codes to simulate the important phenomena
- Performance of tests to obtain the additional data
- Validation of codes with test data.

The NRC will look at core physics to demonstrate safety by way of its impact on fuel performance and source terms (i.e., the primary fission product barrier), fluence and its effect on reactor vessel performance (typically the secondary barrier), and as the source of heat transfer through primary and secondary loops to associated heat exchangers, turbines, and containment (the last FPT barrier). Acceptable performance must be demonstrated during normal operating conditions, anticipated operational occurrences (AOOs), design basis transients, and design basis accidents (DBAs).

For core analysis, developers of neutronics tools for non-LWRs will need engagement with NRC staff to clarify how to adequately characterize and adjust traditional core physics methodologies. Criticality safety analysis must demonstrate safety during fuel manufacture, handling, operation, intermediate storage, and discharge.

The licensing effects analysis related to ART research on analytical safety codes and methods is provided in Table 2. Licensing-oriented recommendations and observations concerning research on this topic are provided in Subsection 4.2 and Section 5.

2.1.4 Core Heat Removal

Testing and confirmation must address all issues regarding core heat removal. This topic is closely related to core design and may involve other plant SSCs relied upon to provide or support heat removal during AOOs, DBAs, and certain beyond design basis events (BDBEs). How these elements relate to safety must be well understood and merged into the plant safety case. Research that supports heat removal analysis becomes more important as a technology supplier pursues the use of more passive methods of core heat removal.

The licensing effects analysis related to ART R&D for core heat removal and core design is provided in Table 3. A priority recommendation that affects this topic is provided in Subsection 4.3.

2.1.5 Materials Analysis

A sound technical basis is required concerning the integrity and modes of failure of SSCs important to safety. R&D support is needed for situations using new materials or existing materials in new applications. Time-dependent material failure criteria must be developed to ensure safety is maintained (with margin) and demonstrate satisfactory operational life. Development of common standards concerning material applicability and adequacy may come from trade organizations such as the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code for advanced reactors. Codes developed by consensus organizations like these are recognized by NRC as an effective and acceptable means to confirm structural material design as technically sound and appropriate.

For many non-LWR technologies (particularly those with elevated temperatures or employing new cooling mediums), a materials performance database may not exist to support an assessment and must be developed. Time-dependent failure criteria for materials in high-temperature and/or corrosive environments may become evident through testing planned to expressly target material performance throughout the operational life of a component.

Confirmatory assessment tools and predictive performance models are also an element in materials analysis. Prominent areas of evaluation include initial material behavior before and after fabrication, effects of irradiation on material properties, aging effects in the environment where it is used, and corrosion behaviors of structural materials under varying plant conditions. Lack of operating experience is typical for new reactor types, thereby making this topic a persistent area of licensing concern.

The licensing analysis related to current or planned ART materials research is provided in Table 4. Recommendations and observations related to ART materials science research are provided in Subsection 4.4 and Section 5.

2.1.6 Instrumentation and Control

Developers must ensure instrumentation and control (I&C) systems suited for their designs can adequately measure, diagnose, and respond to normal and off-normal parameters when required. New configurations will trigger new I&C needs and signal a call for new sensor types, updated data integration techniques, and first-of-a-kind human-interface displays. Some I&C sensors may need to operate in environmental conditions significantly different and far harsher than current LWR fleet experience can address. Temperature, pressure, flow, and neutron instrumentation may operate at higher temperatures or under strongly corrosive conditions. New combinations of high radiation, high temperature, and chemically reactive process environments will create formidable challenges to I&C developers regarding instrument functionality, reliability, precision, and maintenance.

Evolutionary capabilities will be needed for in-core monitoring and surveillance diagnostics for key parameters (i.e., power, flow, etc.) in advanced reactor environments, which reduce inherent uncertainties and associated licensing conditions that would otherwise result from use of ex-core detectors or other less-accurate methods. Methods for monitoring performance of passive cooling systems will be needed for heat removal safety systems in many advanced reactor technologies. Techniques and methods for inspecting and verifying the integrity of reactor internals are also a challenge facing advanced rector I&C developers.

A licensing effects analysis regarding ART research activities for I&C systems is provided in Table 5. A licensing-oriented recommendation in this area is provided in Subsection 4.4.

2.1.7 Safeguards and Security

New reactors are expected to integrate greater levels of security and special nuclear material (SNM) safeguards into their basic design. Provisions for the conduct of assessments will be needed to evaluate actual levels of protection afforded by the applied measures. Formal expectations are currently being proposed by NRC in the form of "Preliminary Draft Guidance Non-Light Water Reactor Security Design Considerations."[12]

Development programs are needed at some level to establish effective design security and safeguard measures commensurate to the specific technology. It is notable that due to the unique fuel form associated with liquid-fueled MSRs, a dedicated R&D effort is needed to establish the technical basis for material control and accountability (MC&A). While HTGR, SFR, and solid-fueled MSR technologies will likely look to adapt existing safeguard measures proven effective with LWR technology, salt-fueled MSRs will require a radically new approach to SNM material inventory, accountability, and tracking.

A licensing effects analysis has been performed related to ART research opportunities in safeguards and security. Results are provided in Table 6. A licensing-related observation is provided in Section 5.

2.1.8 Accident Sequences and Initiators

Scenarios that portray and bound normal and off-normal design conditions and the phenomenology associated with those conditions will be evaluated. Extensive R&D to support identification of these representative conditions is needed to provide analysis codes and models that characterize phenomena of interest. Consequently, this topic is closely related to "Analytical Codes and Methods" as described in Subsection 2.1.3.

A long-standing non-LWR licensing concern involves the regulatory process by which event scenarios can be systematically and consistently identified and selected for safety assessments. Once the process is understood, research can be planned and performed to assure information necessary to the conduct of the assessments is generated and available for licensing use.

A licensing effects analysis pertaining to R&D opportunities was performed on an evaluative process now being proposed to NRC. The process will provide a means by which LBEs can be identified and evaluated in a TI-RIPB manner using established probabilistic risk assessment (PRA) methods. As the TI-RIPB process of LBE selection is further refined and finalized, it will be released for use as regulatory guidance. This will give technology developers greater certainty in understanding how advanced reactor accident initiators and sequences can be identified and addressed by R&D.

Table 7 denotes the regulatory effects analysis for Accident Sequences and initiators that includes LBE selection. An observation related to the licensing framework modernization topic is provided in Section 5.

2.1.9 Probabilistic Risk Assessment

The NRC has stated that new reactor designs are to be risk-informed.[1] This makes PRA processes an important component in technology development and evaluation. However, the experience base that forms the foundation of most PRA processes is limited for many non-LWR SSCs. This lack of experience can undermine systems modeling needed for a PRA analysis and call into question underlying hypotheses on how passive systems are treated, the validity of risk metrics that replace traditional core damage frequency and/or large early release as figures of merit, component failure rate data, and (perhaps most importantly) use of materials and systems important to safety. Both the applicant and NRC staff must determine the technical adequacy of a PRA used to support a design and safety decision and know if that assessment adequately supports the risk insights used to justify a consequence prediction.

A licensing effects analysis of certain PRA activities is provided in Table 8. Future R&D on this topic is expected to benefit from a priority recommendation made in Section 4.4 and observations in Section 5 focused on developing a better understanding of the issues and expectations needed when establishing a technology development strategy.

2.1.10 Structural Analysis

While structural analysis methods and tools for large LWR plants are mature, standardized, and benefit from extensive application histories, it is unclear whether those same tools can be used in important non-LWR SSCs without substantial modification. While confirmatory analysis of structures may be possible to varying degrees, it is likely that additional investigations will be necessary for innovative new SSCs. While it is often presumed that the analysis tools currently in use can be satisfactorily applied to non-LWRs, the presumption is questionable with respect to seismic and structural analyses that rely on PRA techniques. Data are needed to both develop refinements and complete a safety review if existing tools are deemed insufficient. For example, non-LWRs that use new key safety components need fragility information if a seismic response analysis is to be considered valid.

The development, qualification, and deployment of seismic isolation (SI) technology at a nuclear facility represents' a major plant safety enhancement opportunity that has attracted the interest of both non-LWR developers and the NRC. Seismic isolation was originally developed for and matured through the building and bridge industries and is now under active consideration for nuclear facilities.

SI technology is attractive because it offers a cost-effective engineering solution addressing multiple challenges faced by advanced reactor suppliers. Seismic isolation can be applied at the foundation level of a facility to isolate an entire structure or at attachment points of large SSCs. Some advanced reactor designs like the modular HTGR are expected to use a deep (below grade) embedment that envelopes the entire reactor core and associated heat exchange systems; SI equipment can be incorporated into these design not only to increase resilience to seismically-induced loads but also expand the range of siting options. Consequently, development of SI equipment and accompanying analysis techniques represent a promising TI opportunity that encourages new deployments, reduces costs, and ensures attainment of seismic safety objectives.

A variety of new seismic analysis tools are needed to support development of integrated SI systems and the assessment of seismic impacts to below-grade SSCs. These new types of analysis tools have yet to be reviewed by NRC and qualified for use at a domestic nuclear facility.

A licensing effects analysis on a pathway enabling use of SI in advanced reactor systems is provided in Table 9. An additional research activity concerning seismic analysis of embedded structures has been analyzed in Table 2, Item 2.f. A future licensing concern is noted in Subsection 5.8.

2.1.11 Human Factors

Advanced reactors create new operational and maintenance challenges substantially different from standard LWR practices. Examples of the type and extent of these variations include the control room, use of computer-based technology as part of an integrated digital I&C program, and modified alarms, controls, and displays that enable reductions in plant operations and maintenance staff size. Potential research considerations in this area include a definition of plant functional requirements and how those functions are allocated on the basis of human-related factors.

No RTDP analysis on human factors has been performed in relation to ART research. Table 10 is reserved for future use on human factors topics. No licensing-related recommendation is provided.

2.2 Quality Assurance

It is critical that advanced reactor R&D be supported by a QA program compliant with requirements recognized by NRC. Applicants are required to submit high quality information in applications for a CP or OL (10 Code of Federal Regulation (CFR) Part 50) or an ESP, DC, and COL (10 CFR Part 52) conforming to methods and administrative controls of 10 CFR, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This requirement assures adequate confidence that SSCs will perform their required safety function when required. These requirements are also applicable to DID equipment and tests and activities that affect non-safety-related SSCs, but support overall safe plant operations.

A nuclear plant R&D QA program will normally be based on American National Standards Institute (ANSI) N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants," and associated daughter standards. Guidelines provided in ASME Standard NQA-1, "Quality Assurance Program for Nuclear Facility Application" (with applicable addenda), as endorsed by RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," identify the specific QA criteria that satisfy 10 CFR, Part 50, Appendix B. Further information on acceptable methods for complying with 10 CFR Part 50, Appendix B provisions is available in:

- RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"
- RG 1.33, "Quality Assurance Program Requirements (Operation)."

The scope of an advanced reactor QA program begins with initial technology development and high-level design activities and continues through final design, facility construction, and eventual operation. Since applicants rely heavily on R&D test data (and associated safety conclusions) that might be based on ART studies, it is important to establish a compliant QA program early in an ART technology development project. A quality assurance program description (QAPD) document is generally expected for ART R&D activities that generate data used to support a design safety case. The QAPD (based on 10 CFR, Part 50, Appendix B) will identify and implement the QA requirements applicable to the research so as to satisfy future licensing requirements.

The NGNP Project at Idaho National Laboratory (INL) developed the NGNP QAPD (PDD-172)[13] that addressed QA requirements set by 10 CFR, Part 50, Appendix B. By doing this, a technically defensible basis for ART R&D was created for use in future advanced reactor licensing actions. This QAPD was developed based on NRC NQA-1-2008/1a-2009, "Quality Assurance Requirements for Nuclear Facility Applications," as provided under RG 1.28.

Subsequently, when the NGNP Project was suspended, PDD-172 was inactivated and superseded by the INL Quality Assurance Description Document, PDD-13000[14], until such time NGNP may be resumed. The INL ART TDO Quality Assurance Program Plan (QAPP). Currently, PLN-2690 documents the implementation of the INL QAPD.[15]

It should be noted that while the NGNP QAPD, the INL QAPD, and current ART TDO QAPP addresses all 18 QA criteria established under 10 CFR, Part 50, Appendix B, NRC review of an early version of the NGNP QAPD determined that not all criteria were applicable. This was because only certain QA program elements were deemed applicable to early technology development of that project.

Fourteen NQA-1 program elements were seen as applicable to ART R&D during NGNP pre-licensing technology development.[16] These were:

- Organization – Establishing the QA organization commensurate with duties and responsibilities.
- Quality Assurance Program – Establishing the necessary measures to implement a QAP in order to ensure that activities are in accordance with governing regulations and license requirements. The QAP applies to those quality-related activities that involve the functions of safety-related SSCs associated with the design, fabrication, construction, and testing, as well as managerial and administrative controls to be used, to assure compliance with applicable regulatory requirements. Examples of safety-related activities include, but are not limited to, basic, applied, and developmental research, determination of SSC safety class, engineering related to safety-related SSCs, geotechnical investigations, engineering analysis, seismic analysis, meteorological analysis, and document control.
- Design Control – Establishing the necessary measures to control the design, design verification, and analysis activities of safety-related items and services. The design process includes provisions to control design inputs, outputs, changes, interfaces, records, and organizational interfaces.
- Procurement Document Control – Establishing the necessary administrative controls to ensure that applicable regulatory, technical, and QA requirements are included or referenced in procurement documents.
- Instructions, Procedures, and Drawings – Establishing the necessary measures and governing procedures to ensure that activities affecting quality are prescribed by, and performed in accordance with, documented instructions, procedures, or drawings of a type appropriate to the circumstances and which, where applicable, include quantitative or qualitative acceptance criteria.
- Document Control – Establishing the necessary measures and governing procedures to control the preparation, review, approval, issuance of, and changes to documents that specify quality requirements or prescribed how activities affecting quality, including organizational interfaces, are controlled.
- Control of Purchased Material, Equipment, and Services – Establishing the necessary measures and governing procedures to control the procurement of items and services to ensure conformance with specified requirements.
- Inspection – Establishing the necessary measures and governing procedures to implement inspections that assure items, services, and governing procedures to implement inspections that assure items, services, and activities affecting safety meet established requirements and conform to applicable documented specifications, instructions, procedures, and design documents.
- Test Control – Establishing the necessary measures and governing procedures to demonstrate that items subject to QA provisions will perform satisfactorily in service. This includes applicable procedures that include: (1) instructions and prerequisites to perform the tests; (2) the use of proper test equipment; (3) acceptance criteria; and (4) mandatory verification points as necessary to confirm satisfactory test completion.
- Control of Measuring and Test Equipment – Establishing the necessary measures and governing procedures to control the calibration, maintenance, and use of measuring and test (M&TE), which provides data to verify that acceptance criteria are met.

- Nonconforming Materials, Parts, or Components – Establishing the necessary measures and governing procedures to control items, including services that do not conform to specified requirements, in order to prevent inadvertent use. Controls provide for identification, documentation, evaluation, segregation, disposition of nonconforming items, and notification to affected organizations.
- Corrective Action – Establishing the necessary measures and governing procedures to promptly identify, control, document, classify, and correct conditions adverse to quality.
- Quality Assurance Records – Establishing the necessary measures to ensure that sufficient records of items and activities affecting quality are developed, reviewed, approved, issued, and revised to reflect completed work.
- Audits – Establishing the necessary measures and governing procedures to implement audits to verify that activities covered by the QA program are performed in conformance with established requirements.

Four QA requirements were deemed by NRC as not applicable during modular HTGR technology R&D and high-level design activities. These were:

- Identification and Control of Materials, Parts, and Components
- Control of Special Processes
- Handling, Storage, and Shipping
- Inspection, Test, and Operating Status.

NRC staff also noted that either a supplemented QAPD should be submitted if the scope of the NGNP at that time were expanded to include design and/or construction activities in accordance with becoming an actual applicant under 10 CFR Part 52. Alternatively, any future applicant or licensee planning to design and/or construct a NGNP-type reactor based on NGNP research and development efforts should submit an independent QAPD covering the appropriate scope of activities in accordance with quality assurance regulations and guidance in place at that time.

2.3 GAIN

DOE-NE recently launched a public-private relationship to better organize relevant DOE programs that help address cost and time-to-market challenges associated with advanced reactor R&D. The Gateway for Accelerated Innovation in Nuclear (GAIN) is an important RTDP concern because it greatly increases opportunities to access key technical, regulatory, and financial resources that nuclear energy innovators need for commercialization. It does this by offering a single point of contact for accessing:

- Government-owned nuclear and radiological experimental facilities and related systems test capabilities on topics such as thermal-hydraulic loops, control systems, etc.
- Computational capabilities and state-of-the-art modeling and simulation tools.
- Data and technology support information at knowledge and validation centers.
- Site information and use options suited to technology demonstration.

GAIN assures advanced reactor technologies at a low technology readiness level (TRL) have shared access to DOE capabilities in R&D in materials testing, analysis, modeling, code development, etc. The needs of more mature technologies generally trend towards larger and more expensive experimental programs and test platforms that are focused on component evaluations and demonstrations. The DOE national laboratory infrastructure is essential to the performance of many projects and programs.

The RTDP supports GAIN objectives by helping assure the "licensing readiness level" of non-LWR technologies is evaluated and communicated commensurate with TRL advancements. It does this by identifying high-priority ART R&D activities related to the advanced reactor safety basis/licensing technical requirements and providing recommendations focused on addressing regulatory criteria.

3. ART REGULATORY TECHNOLOGY DEVELOPMENT PLAN

Contemporary ART R&D priorities led the RTDP analysis to focus on activities and opportunities associated with three classes of advanced reactor technologies, i.e., the modular HTGR, liquid-metal cooled SFR, and a generic type known as MSRs. A variety of planning documents related to these concepts are available for review.[3,5,6,7,17,18,19,20,21,22,23,24,25,26,27,28,29,30,31]

Additional information was collected through discussions with ART topical area research leads and subject matter experts to confirm activity descriptions, status, and bring planning information up-to-date where indicated. That information was binned according to topic and entered into a tabular format for licensing impact evaluation. Actual evaluations were performed by ART Licensing staff located at the INL and relied heavily on experience gained during and following the NGNP project. The regulatory effects analysis sought to highlight long-term research needs essential to licensing success; analysis results are documented in Tables 1-10 of Section 3 with specific licensing priority recommendations presented in Section 4 and licensing-related observations and concerns noted in Section 5.

The following outlines the strategy, evaluation criteria, and ranking protocol employed in the licensing effects analysis.

3.1 Regulatory Importance

Once a research opportunity has been identified, a brief description was developed that relates how R&D activity results (i.e., the data generated from test plan completion) might support a technology safety case and licensing requirements. The "Regulatory Importance" of a research activity examines the role the study is expected to play in support of licensing. A regulatory importance ranking is assigned to the activity as follows:

- *High* – A phenomena or topic that is of first order (fundamental) importance to design safety and a critical component of the independent safety review. Information generated by the research activity is understood to be essential in successfully meeting safety criteria.
- *Medium* – A phenomena or topic of secondary (contributing) importance to the design, safety case, and safety review process. Alternative regulatory options may be available to address the issue under review. The issue is made more important due to factors such as addressing concerns important to multiple advanced reactor technology types or significantly influencing the timeline for commercial deployment. While research with a "Medium" level of regulatory importance is generally less imperative than an item with "High" regulatory importance, completion of the activity is still considered essential for licensing.
- *Low* – A phenomena or topic not currently considered significant to the design safety case or essential to support the independent safety review process. These items represent a low level of contemporary licensing concern.

Rankings are assigned based on a general consensus opinion of ART Licensing personnel experienced in the regulatory framework and licensing environment.

3.2 State of Knowledge

Establishing a licensing priority for DOE-sponsored R&D must consider the current state of topical knowledge and information "gaps" that must be addressed for licensing success. Technology development research priorities trend higher when inadequate knowledge exists to resolve a safety issue. Ranking guidance on the "State of Knowledge" criteria are identified as:

- *High* – A physics-based or correlation-based modeling capability is available that represents the phenomenon or issue over parameters of interest (with adequate margin). A body of data is available that likely satisfies regulatory quality assurance requirements and can be used to validate and predict capabilities and/or can support further development to completion (which, in this context, includes NRC review and acceptance of the capability).
- *Medium* – A candidate model or appropriate means of correlation has been identified and is available to address most phenomena or issues of concern over a considerable portion of the parameter envelope. Supporting data are available, but the database is not necessarily complete or contains elements of questionable quality. Only moderately reliable system capability assessments are allowed by this state of knowledge.
- *Low* – Functional models or predictive capabilities are uncertain, speculative, or do not exist. Existing databases are insufficient to reliably support safety assessments due to high levels of uncertainty.

Rankings are assigned based on information gathered from various research plans and informed opinions communicated to licensing staff by ART technical area research leads and subject matter experts.

3.3 Status of Research

Establishing a licensing priority recommendation for R&D activities requires an understanding of the state of development of necessary knowledge. To properly frame a R&D priority with respect to licensing importance, key information gaps may be identified and planned for resolution on a timeline that is or is not conducive to an established licensing timeframe. In other words, once the status of a critical research activity is characterized, a time-phased work sequence is considered to ascertain if major predecessor/successor relationships exist between individual research activities that might adversely impact the licensing "critical path" timeline. Research status becomes particularly significant for activities with very long lead times that are sequentially dependent upon completion of other related research (such as doing in-core fuel irradiations prior to conducting PIE and safety "heat-up" tests, which in turn generate data essential for MST analysis code development).

Research activity status is not discretely ranked in the RTDP, but rather focuses on capturing time-related information pertinent to data needs. Related questions typically consist of:

- Is essential research already planned, underway, and adequately resourced? If so, licensing priority is reduced in recognition of pending issue resolution.
- Do predecessor/successor relationships exist that can adversely affect research planning and performance? If so, licensing priority is increased due to the influence of sequential test plan completion on the licensing critical path.

- Does the research address an essential concern in establishing or assessing the safety case? Are other options available to address the need? Licensing priority can be reduced if completion of a test plan is not essential to establish the safety case.
- Are suitable testing capabilities, essential infrastructure, and technical support available to the investigation? If not, what are precursors to the study? Licensing priority may increase if a testing infrastructure or capability is inadequate or unavailable to support the R&D.
- Is there a long lead time (i.e., >5 years) associated with activity completion? Licensing priority is generally reduced if a test plan can be completed in a relatively short period of time.

Information on research status was collected primarily from ART research plans and updated according to inputs provided by technical research area leaders and subject matter experts within ART.

3.4 Licensing Priority Evaluation

Using the ranking pattern described in Subsections 3.1 through 3.3, a foundation is developed for performing activity-specific licensing priority evaluations. Interpretation of information was done by ART Licensing personnel and influenced by regulatory perspectives and experiences gained through recent NRC interactions. The result is a simple preliminarily assessment system that projects expected licensing levels of concern on discrete units of ART research. By capturing summary statements on anticipated regulatory impact, state of knowledge, and status of research, licensing evaluations and priorities are derived for research opportunities planned and/or underway.

A four-increment "licensing priority" scheme is established to convey analysis results. Guidance on increment ranking values consists of:

High – A licensing priority that suggests the research activity results is expected to address a major safety concern important to future licensing success. Research activities with this priority generally exhibit a high or medium level of regulatory importance, current technical knowledge is low to medium with respect safety case development and NRC informational needs. A long lead time is expected to generate validated test results.

A "High" licensing priority designates the highest level of licensing concern relative to the activity under review.

Medium – This priority denotes R&D that has a high or medium level of regulatory importance, the state of necessary knowledge ranges from low to medium, but research plan completion is not expected to be excessively lengthy with respect to licensing timelines. This rating may include extensive research programs very important to regulatory decisions but the activity is already planned and adequately resourced. Research activities with this rating are acceptably scheduled according to the understood licensing critical path.

A "Medium" priority denotes R&D that is significant to plant safety, but present minimal risk to licensing schedules. Mitigating factors for the classification include short lead times for test plan completion and a need for design-specific information from suppliers to support test planning. A "Medium" priority can also identify "watch list" research that may become a higher future licensing concern.

Low – This priority characterizes R&D activities with a medium or low level of regulatory importance, the state of knowledge is low to high with respect to anticipated regulatory needs, and the activity does not have a long lead time to complete.

Research activities with a "Low" priority may be a necessary component in reactor technology deployment but test results will generally have a low-level of influence in licensing decisions. "Low" priority research items are not expected to challenge the critical path for license application development or an independent safety review.

None – Denotes research plans and activities not otherwise designated as "High, Medium, or Low."

An activity with a "None" priority is not considered an ART research priority with respect to licensing and is therefore outside the nominal scope of RTDP recommendation.

It should be noted that the aforementioned structure is not a rigid evaluative metric. Instead, the flexibility of the process allows for adjustments in response to unique factors that may not be specifically addressed in Subsection 3.1 thorough 3.3 criteria.

3.5 ART Research Activities

Table 1 through 10 identify and evaluate ART R&D opportunities with respect to licensing significance. The tables communicate specific issues, factors, and concerns relative to ART research that are pertinent to establishing a regulatory safety case and completing an independent safety review. Information in each table is partitioned according to reactor technology (e.g., HTGR, SFR, MSR).

Section 4 summarizes consolidated conclusions and recommendations derived from information appearing in Table 1 through 10. Section 5 identifies additional observations likely to become a future licensing priority.

Description

Fuel Qualification:

Depending on technology attributes, information relating to the design, manufacture, and use of fuel in support of safety may be a major concern when identifying and scheduling ART R&D. FQ programs usually require lengthy irradiation tests and PIE to evaluate new or modified fuels different than those currently used in LWRs. Because very long lead times and critical infrastructure support is necessary to support a comprehensive in-core irradiation and PIE campaign, the scope of outstanding fuels research needs is almost always a high-priority licensing concern. Qualification of new fuel is an activity demanding specialized resources and long-term planning. Thus, repetitive testing should be minimized and tests planning done to reduce uncertainty in resulting data to the maximum possible extent.

Appropriate analytical tools and a robust supporting experimental database are needed to support analysis of fuel system response to anticipated ranges of conditions. Simulating fuel performance and FPT, retention, and release into the environment under accident conditions is also a direct function of advanced reactor design.

Mechanistic Source Term:

A mechanistic approach to source terms development will be used when establishing the technical basis for a subsequent safety analysis. This involves allocating appropriate credit to SSCs for radionuclide retention capabilities. The safety approaches that are now being proposed should be consistent with the presence of multiple barriers in radionuclide transport to the environment. Multiple barriers to radionuclide release are a basic expectation of the regulatory environment. A MST evaluation must be based on detailed analysis of fuel and reactor systems behavior during normal operations and bounding accident conditions. Source terms developed with a mechanistic approach must identify and characterize radionuclide inventories that exist elsewhere in the facility as well as that arising from the core itself. MSTs can be used for other purposes such as equipment environmental qualification, control room habitability analyses, and assessments of severe accident risk.

NOTE: The R&D associated with MST development is reliant upon FQ research. The performance data associated with fuel type and core design provide the first step in analytical MST modeling. Therefore, the licensing effects analysis of ART MST research is done in combination with the FQ analysis.

Table 1. ART research regarding fuel qualification and mechanistic source term.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	**Establish fuel service conditions and performance requirements for normal and off-normal operations**								
1.a	HTGR	Define fuel service conditions for normal operation; supplement by peak fuel temperature, burnup, fluence, burnup from fissions of bred plutonium, maximum time-at-temperature and postulated accident conditions.[a]	High	This activity generates essential information that interfaces with LBE selection methods, accident analysis, and consequence predictions. It is also associated with fuel qualification and MST development that are fundamental to safety review evaluations.	High	TRISO-coated particle fuel service conditions are being addressed by Advanced Gas Reactor (AGR) Fuel Program irradiation tests. The tests are based on the most stringent performance requirements for two different core types of HTGRs—prismatic block core and pebble-bed core. Normal conditions are based on best available conservative code predictions for coated particle fuel while accident conditions will be derived from best available information on the nuclear, thermal, and chemical environments predicted during anticipated LBEs for a preliminary modular HTGR design.	AGR test regimes addressing this issue are underway at INL.[b] Post irradiation tests on AGR-1 and AGR-2 test fuel provided sufficient fuel failure rate data to support initial conclusions about fuel accident performance. Work remains on how laboratory data can be up-scaled to represent industry produced fuel. AGR-3/4 irradiations are concluded and PIE/safety testing is underway. AGR-5/6/7 irradiations are planned for completion in 2020 and PIE/safety testing finish in 2024. Supplemental tests and additional verifications may be needed to support pebble fuels once fuel service conditions are defined in conjunction with the final design.[c]	Medium	Establishing normal and off-normal fuel service conditions is a fundamental licensing issue being generically addressed in the AGR program. Issues like research data scale-up and additional testing for pebble fuels will likely be part of that design certification and addressed during license application development. Completion of AGR testing is a critical HTGR licensing concern but licensing priority is reduced because AGR testing is roughly 2/3 done and proceeding on schedule that supports planned deployment. Verification of AGR findings against final design conditions will be a future licensing concern for the applicant.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.b	HTGR	Determine how varied combinations of fuel operating parameter values (e.g., maximum fluence with moderate burnup, moderate fluence with maximum burnup, low operating temperature with maximum fluence), affect fuel performance in HTGR operating and accident conditions.[a]	High	This activity factors into critical regulatory and technical adequacy assessments concerning reliance of safety decisions on the accelerated AGR test irradiations conducted at the INL Advanced Test Reactor (ATR) to address the higher ranges of fuel operating temperature, burnup, and fluence.	Medium	The AGR program has yet to see evidence of significant parameter path dependence during AGR testing for normal fuel operating conditions. However, these determinations are contingent upon HTGR core design estimates (yet to be finalized). Further evaluation of this issue is seen by NRC as a possible necessity.[a]	Informational gaps in “path dependence” coverage is considered in AGR test 5/6/7 planning (PIE/safety testing scheduled for completion in 2024).[d] Additional irradiations dealing with fission product transport code validation (i.e., AGR-8) is indefinitely delayed indefinitely pending availability of design-specific information required to refine experimental protocols. When completed, AGR test data will be evaluated for use significance in validated phenomenological models of TRISO fuel performance under operating and accident conditions.	Medium	Path dependence variance is an NRC staff concern documented during NGNP pre-licensing interactions.[a] Evaluations of fuel operating parameter path dependence will be an indicator of robustness in AGR test results. The licensing concern is “medium” because topic resolution is planned under the current AGR test program. Future assessment will be needed to assure collected AGR data adequately represents performance in the final design.
1.c	HTGR	Identify the substantial uncertainties and undetected anomalies in fuel normal operating service conditions. Key parameters should include maximum normal fuel operating temperature.[a]	Medium	Timely R&D can clarify needs and circumstances surround development and qualification of advanced HTGR sensor systems needed in prototype monitoring, surveillance, and testing programs. Without supportive information to resolve safety uncertainties, unnecessarily restrictive conditions and requirements may be added to a HTGR COL.	Medium	During NGNP pre-licensing interactions, NRC staff documented an understanding that fuel testing uncertainties will likely trigger additional requirements concerning verification of initial and normal fuel operating conditions. This could be performed through a special operational monitoring, testing, surveillance, and inspection program at the (first) technology demonstration reactor module.[a]	DOE/INL believes the AGR Irradiated Fuel Test #7 plan will demonstrate sufficient margins to failure for the TRISO fuel form under normal operating and potential accident conditions.[b] It is expected that testing will experimentally address the most significant uncertainties.[d] However, NRC staff has stated a belief that HTGR core analysis and core monitoring issues can only be partially addressed by analytical means and that separate effects validation tests will likely be needed.[a]	Low	A precise description and thorough understanding of in-core monitoring options and initial power ascension tests remains to be established with initial HTGR reactor module licensing. Specific conditions remain to be defined and accepted by the NRC. AGR test results should be incorporated into this analysis as much as possible. Commitments leading to final resolution of this NRC concern are to be established by the applicant early in license application development.
1.d	HTGR	Qualify UCO-based TRISO fuel by engaging an NRC review of: 1) Fuel design characteristics; 2) Fuel product specifications; 3) A description of the fuel fabrication process; 4) Statistical QA methods that assure specifications are met; 5) Irradiation behavior of TRISO fuel (in-pile performance and PIE); 6) Fuel safety test results; and 7) The establishment of a fuel performance envelope with failure rates for normal and off-normal conditions.	High	TRISO-coated particle fuel performance is fundamental to the HTGR safety basis, but a significant source of licensing concern. Since FQ is a specialized, long lead R&D activity without well-defined “top level” regulatory criteria against which success can be measured, a preliminary “limited scope” TRISO FQ effort could address to-date testing and confirm acceptability of initial test conclusions and facilitate the identification of additional concerns that may need R&D in order to achieve full TRISO-coated particle FQ.	High	Based on information now available through the AGR program, a generic limited-scope FQ submission can be made on UCO FQ that examines: 1) TRISO UCO fuel design characteristics with rationales; 2) TRICO UCO fuel product specifications; 3) A description of fuel fabrication processes; and 4) Statistical QA methods that assure specifications are met. Remaining FQ issues dealing with fuel irradiation behaviors, safety test results, and fully defining the fuel performance envelop in relation to specific design attributes can be addressed in later supplements after AGR 5/6/7 PIE and safety testing are completed (i.e., 2024).	Conclusions of AGR testing done to-date can support a limited-scope TRISO FQ report that generically: 1) Confirms fuel design service conditions are appropriate; 2) Confirms fuel quality and safety performance criteria are appropriate; 3) Confirms selected process specifications are appropriate to FQ testing; 4) Confirms testing and inspection methods for fuel fabrication are necessary and sufficient for key parameters; and 5) Confirms sampling and statistical analysis methods proposed for acceptance and product specification compliance are appropriate. Technical expertise concerning AGR testing is available now to support FQ report development and NRC interactions.	Medium	The AGR program will generate additional safety test data for particle FQ (i.e., AGR 5/6/7). These could be combined with design information to later complete a full TRISO FQ package. While applicants are ultimately responsible for qualifying fuel used in their design, assistance on major (generic) portions of the FQ basis is available now through the AGR project and fuel fabricator (e.g., BWXT). This “staged” review would significantly reduce licensing uncertainty; priority is “medium” for the licensing risk reduction offered by this option.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.e	SFR	Evaluate SFR fuel acceptance criteria for normal and off-normal operations (e.g., core disruptive damage functions, cladding thermal creep strain limit) Identify sources of major uncertainty (e.g., burnup, fluence, thermo-physical properties) and define how they may influence key parameters of interest (e.g., fuel and cladding temperatures).[e]	High	Understanding fuel behaviors and parameters that influence fuel performance during steady-state irradiations and transient conditions (including anticipated operational occurrences and postulated accidents) are important to FQ and the selection and analysis of LBEs. Understanding the modes of potential fuel failure is critical to enabling a MST assessment.	Medium	Pending confirmation through design comparison, sufficient information is believed likely to be available from legacy DOE EBR-II and FFTF operations to support most key SFR FQ/licensing issues provided that design remain within the historic experience basis of metallic fuel.[f] These data include up to 10% burnup, peak cladding temperature of 600°C or less, peak displacement per atom (dpa) of 100, and use un-reprocessed fuel.[e] Acceptable margins in the experience base includes up to 20% burnup, 650°C peak cladding temperature, and variations in fuel pin dimensions.[g] The likely need for additional irradiation testing increases significantly as a proposed design moves away from these parameters.	A SFR metal fuel irradiation testing and physics analysis database has been developed at Argonne National Laboratory (ANL) under DOE-NE's ART program; this data may be adequate (but not necessarily optimal) to support initial licensing provided the design generally conforms to EBR-II.[e,f] While adequacy remains to be confirmed through a formal design review, uncertainties in existing metal fuels data will increase for design event spectrums that move away from historic parameters. Deviations arising from the use of advanced alloys for cladding materials, different fuel pin dimensions, and higher burnups, can be expected to arise in the design basis comparison.[g] Depending on importance to safety, these variations may trigger additional FQ activities that include irradiations and safety fuel testing; such testing will offer opportunity to extend operational envelopes, enable next generation cladding, allow examination of alternative fuel material (e.g., carbides, nitrides, UZrH, carmets), and enable evolution in fuel design.[f]	High	Early interaction between SFR suppliers, NRC staff, and ANL personnel are needed to assure that critical FQ gaps are identified and resolutions planned to complete the experience base. As of 2017, these interactions are starting through industry stakeholders, such as OKLO. If it is found that additional FQ irradiation/PIE/safety testing is needed, testing capabilities must be developed as the currently applicable domestic infrastructure is largely limited to transient testing (i.e., the Transient Reactor Test Facility [TREAT] facility). While creating a new fast reactor test platform is a long-lead time activity, using this testing capability could reduce uncertainties, advance efficiencies, and improve designs. The issue is a high licensing priority due to the tremendous challenges of bringing a new irradiation testing platform on-line.
1.f	SFR	Assess data quality and configuration control standards associated with the legacy EBR-II metal fuel irradiation testing and physics analysis database. The database is being developed under DOE-NE's ART program and is an essential resource in support of forthcoming SFR technology.	High	The legacy fuels database from EBR-II and FFTF are extensive and are supplemented by information from incidents at SRE and Fermie 1.[f] These data are critical to SFR applications and will be used by NRC in FQ and plant safety analysis. Early regulatory interactions concerning historic data quality acceptance and completeness assessment of data coverage, along with implementing of modern configuration controls on the database itself, are essential to meet regulatory data management objectives for safety information relating to normal and off-normal operations, irradiation experiments, and safety tests.	Medium	Domestic SFR fuel knowledge is heavily predicated on historic EBR-II (metallic fuel) and FFTF (mixed oxide fuel) experience. Assessing the state of knowledge with respect to licensing need requires design-specific information from suppliers and confirmation from regulators that these data are of acceptable quality, coverage, and applicability for use in licensing a modern SFR.[g] Near-term domestic reactor suppliers will use metallic fuel, thus making FQ information based on DOE's EBR-II facility the leading priority. Salvaged mixed oxide fuel data from FFTF will also be needed by applicants using those fuel types, but on a longer timeframe.	A searchable SFR metal fuel irradiation and physics analysis database has been established at ANL; NQA-1 data quality level evaluations have been planned and the data has been placed under NQA-1 recognized configuration control protocols.[i] A SFR oxide fuel testing database has yet to be established at Pacific Northwestern National Laboratory (PNNL) covering oxide fuels; oxide fuel quality determination evaluations have yet to be planned.	High	EBR-II data is essential for metallic FQ purposes and crucial to SFR licensing success. A NQA-1 assessment of EBR-II (and later FFTF) information and a completeness survey with design-specific data needs are precursors to identifying gaps and additional fuel research planning to address gaps. Since fast reactor fuel research is a complex, long lead time activity with severely limited infrastructure support, a high licensing priority is given this activity. Confirming NRC acceptance of legacy data quality for safety review purposes is a top near-term licensing issue.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.g	MSR	Define "Fuel Qualification" in terms applicable to MSR technology. From that, develop fuel acceptance criteria and service conditions for normal and off-normal conditions. Evaluate quality of historic test data used to develop the criteria and characterize sources and magnitude of uncertainty. Plan FQ safety tests to fill technical gaps identified across the LBE spectrum.	High	Fuel qualification focuses on assuring safe fuel performance (with margin) during all LBE conditions. Thus, establishing fuel acceptance requirements starts by defining the role a fuel plays in the overall plant safety case. Lack of well-defined "top level" regulatory criteria on FQ also contributes uncertainties. Since the role molten fuel plays in safety may vary significantly by design type, all fuel data evaluations and R&D test planning should start with a practical definition about what qualified molten salt fuel looks like both initially and as the fuel changes over time and through use.	Low	The ORNL MSR-E test reactor produced a wealth of data useful to develop fluoride MSR design approaches. However, these data are insufficient to address contemporary licensing requirements. Defining and addressing gaps starts by understanding how the fuel affects safety and consideration of a paradigm that may entail movement away from thinking of fuel as a function of "barriers" in fission product release control to perceptions built on buildup, cleanup, precipitation, retention, and release parameters. Understandings should be derived with advisement from NRC with reviews starting at a high level followed by technical test reports focused on specific subtopics. Some unique R&D needs already recognized for heterogeneous liquid fuels include delayed neutron moderation both inside and outside of core, salt changes over life and with respect to position, and processing/cleanups during use.[j]	There is no ART fuel R&D currently underway for MSR technology.[k] While discussions are starting in the DOE MSR Technology Working Group concerning qualification options of historic MSR-E data, subsequent efforts that include R&D FQ planning must be refined to accommodate the type of fuel used (i.e., uranium or thorium), whether the fuel will operate in a thermal, epithermal, or fast neutron environment, what coolant is employed, and the role the fuel itself plays in assuring safety.	High	MSR-E produced considerable data suited to design development, but is only a starting point for a safety analysis. This legacy data becomes less useful the more a design move away from the MSR-E basis for safety. It is highly probable that irradiation safety testing will be necessary, especially for fuels used in a chloride salt design. FQ safety testing is a very long lead time activity with a potential for limited or non-existent infrastructure support. Because basic understanding remains to be established on the safety basis and associated molten salt fuel service conditions and performance requirements, a high licensing priority is attached to starting this research activity.
2	**Demonstrate fuel performance requirements are met at normal operating conditions using irradiated fuel at design conditions, fuel irradiation performance monitoring, and post-irradiation examinations**								
2.a	HTGR	Perform irradiation, safety testing, and PIE of UCO and UO_2 TRISO fuel from laboratory- and prototypic-scale equipment should be used to establish normal operation conditions based on performance data.	High	This research broadens options and enhances prospects for meeting necessary TRISO fuel performance requirements. It supports development of fundamental understanding concerning relationship between the fuel fabrication process, as-fabricated fuel properties and normal operation/accident condition performance.[b]	Medium	AGR fuel test program testing is underway. When completed, test results will provide necessary irradiated fuel performance data and irradiated fuel samples for safety testing and PIE concerning key fuel product and process variants.[b]	With AGR test 2 PIE (scheduled for completion by 2020), laboratory-scale irradiation performance testing is concluded. Prototypic-scale testing (with performance margin) will be accomplished through AGR Tests 5/6 PIE/safety testing.[d]	Medium	This research is important to licensing success, but has a good state of overall current knowledge. AGR Test 2 PIE is underway and prototype AGR Test 5/6 PIE is scheduled to confirm information needed for licensing. Licensing priority reduced to "medium" because essential research is underway and tracking towards completion on a timeline that supports deployment schedules.
2.b	HTGR	Perform irradiations, PIE, and safety testing of representative fuel containing designed-to-fail (DTF) particles in support of fission product transport model development.[b]	High	This topic provides information essential to MST development in all designs using TRISO-coated particle fuel. The MST is in turn used to evaluate nuclear safety and risks to the public during the licensing process.	Medium	In-pile gas release data, PIE, and safety test information on fission gas and metal released from particle fuel kernels support the development and refinement of improved fission product transport models essential to an independent safety evaluation. AGR tests will also provide irradiated fuel performance data on fission product gas release from failed particles.[b]	The AGR fuel program will assess and document the effect of impurities on intact and DTF particle fuel performance and related fission product transport. The AGR irradiation tests needed to acquire this information are complete and PIE activities are underway. ART activities needed to address this topic are on track for completion.	Medium	R&D on this topic is a necessary component of the design safety analysis and on track for completion. The topic has a good state of existing knowledge and to-date AGR test results may be sufficient to provide confirmations. Licensing priority is reduced because supporting research is well underway and supports the licensing timeline.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.c	HTGR	Demonstrate the adequacy and representativeness of accelerated TRISO-coated particle fuel irradiation testing.[a]	Medium	Lack of TRISO fuel performance data obtained in a (real-time) modular HTGR neutronic environment has been documented as a concern by NRC staff.[a] Full prototypic data may be needed to demonstrate and confirm (to NRCs satisfaction) that fuel performance is adequately understood and can be predictively modeled (with proposed analytical tools) respective to fuel radionuclide retention and transport.	High	AGR PIE and safety tests intend to provide a broad spectrum of data on TRISO fuel performance and fission product transport within fuel particles, compacts, and graphite materials representative of fuel element blocks. (Although additional future testing will be needed for pebble-type fuel).[c] These data, in combination with in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios) will demonstrate compliance with fuel performance requirements and support development and validation of computer codes.[b]	Multiple AGR tests were planned to provide sample materials and data representative of a full scale design and support TRISO fuel qualification.[b] Based on current AGR test plans and R&D results collected to-date, the need for additional research on this issue (i.e., supplemental to the AGR program) is not considered essential to thoroughly characterize the fuel and demonstrate acceptable performance.[d]	Low	The AGR Fuel program tests are designed to characterize fuel safety using accelerated thermal neutron test conditions. The test plan has been reviewed by NRC and adjusted in response to feedback. While later comparisons may be needed to support formulated conclusions, further ART R&D planning on the issue is not a significant licensing concern.
2.d	HTGR	Evaluate plutonium generation and burnup in TRISO-coated particle fuel test irradiations.[a]	Low	Plutonium burnup is a normal operating service condition parameter that is to be specified for particle fuel. While DOE/INL believes this issue has little effect on TRISO fuel performance, NRC staff documented this as an ongoing concern in the course of NGNP pre-licensing discussions.[a]	Medium	Further research on this topic appears irrelevant to the pebble bed particle fuel design.[d] For prismatic designs, DOE/INL's current approach is to increase plutonium burnup in AGR irradiation tests and rely on neutron absorbers in the test rig to effectively harden the thermal spectrum by reducing the neutron flux in the lower range of the ATR thermal energy spectrum.	Planned AGR irradiation tests/PIE/safety tests will provide sample materials and data that further existing knowledge on this issue. Details of the test program are described in the AGR Fuel Development and Qualification program.[b]	Low	NRC has indicated a desire for a more thorough understanding of plutonium generation parameters in TRISO fuel. Completion of planned AGR tests should address the issue adequately in prismatic block core HTGRs; this is a minor concern for pebble bed core HTGRs. Discussion with NRC should resume once AGR test data becomes available.
2.e	SFR	Ensure existing SFR fuels irradiation safety testing and PIE are available to support fuel design safety goals. If support data are needed outside the bounds of the existing experience base (i.e., substantial deviations in pin dimensions, fuel compositions, higher burnup, etc.), perform additional testing.	High	Fuel test information must be available for review that covers the full spectrum of design basis event (DBE) and DBA. It is currently assumed that sufficient historic data exists to support a basic regulatory safety review of SFR fuel designs that are comparable to EBR-II.[f,g] However, this presumption must be confirmed through a formal design review and additional fuels testing research may be desirable to broaden design options, increase assurances that fuel design performance requirements are being robustly met, and better understand relationships between the fuel fabrication process, fuel properties, and resulting fuel performance under normal and off-normal operational conditions.	High	Over 150,000 metal fuel pins were irradiated up to 20% burn-up without failure in EBR-II. About 1000 taller metallic ternary fuel pins were irradiated up to 15% burnup in FFTF. Fuel reprocessing for 35,000 metal fuel pins was also demonstrated in EBR-II. FFTF oxide fuel irradiation experience covered 48,000 driver pins and over 16,000 test pins up to 20% burnup. The existing SFR fuels irradiation data is considered sufficient for most key regulatory evaluations, but the exact scope of safety review remains to be presented to and confirmed by regulators. DOE-NE's Fuel Cycle Technologies (FCT) program is working on fission product and minor actinide carryover fuel characterization in more advanced fuels. However, this information remains to be understood in relation to the design approaches of prospective technology suppliers.[k]	ART programs are currently focused on knowledge preservation regarding the EBR-II metal fuel irradiation test and physics analysis database. An effort to develop an FFTF fuels irradiation test database is anticipated, but not yet started. The sufficiency of existing legacy data and information for contemporary licensing use must be confirmed.[f,g] Should existing data contain ambiguous or incomplete information on data key to regulatory safety assessments, additional SFR fuels testing may be necessary. Transient testing may be done at TREAT when that facility resumes operations.[f] Other fast spectrum irradiation tests will be a long lead time activity (if required) due to limited infrastructure support.	Medium	Understanding and predicting fuel performance during DBE is a license issue. Based on current understanding, a sufficient experience base may exist from past SFR operational history to licensing the first unit, but regulatory acceptance of this information, the extent of data coverage with respect to the design. Data needs for safety evaluations remain to be confirmed. Presuming existing data is sufficient (but likely are less then optimal) to address basic fuel service and performance requirements and that the fuel can be qualified for use, this activity is treated as a moderate licensing concern for designs similar to EBR-II; other designs departing from EBR-II merit a higher priority concern.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2f	MSR	Ensure MSR fuels irradiation safety testing and PIE information is available to support and verify fuel design safety goals. If data are needed outside the existing experience base, perform additional testing as required to address gaps across the LBE spectrum.	High	Fuel safety test information gathered through representative irradiations must be available during safety reviews to predict fuel performance over the full spectrum of normal operating conditions applicable to that design. It is understood that existing fuels data may be applicable to certain MSR designs (i.e., TRISO-coated particle fuel used in FHRs) but all MSR fuels remain to be confirmed for use in a commercial reactor; supplemental investigative and confirmatory testing will likely be needed and may be used to broaden design options, increase assurance that fuel design performance requirements are robustly met, and better understand relationships between fuel fabrication processes, fuel properties, and fuel performance under all normal operational conditions including shutdown.	Low	Traditional understanding concerning reactor fuel performance does not necessarily hold true for MSRs. Many molten salt fuel designs now proposed must rethink fundamentals of fuel qualification, performance, and safety to move away from terms based on physical fission product release "barriers" towards control regimes based on parameters of buildup, cleanup, precipitation, retention, and release as a function of coolant salt chemistry variations. A licensing focus would remain on fuel system safety, but that safety would consider different attributes like system lifecycles of radioactive material, physical retention and release mechanisms, potentially rapid changes in MSTs used in safety analysis, and how online refueling and/or salt processing might mitigate safety concerns and potential accident consequences.[l] Answers to these questions must be supported by predictions and demonstrations covering the spectrum of normal MSR operations. In general, these tests remain to be planned and performed.	With the partial exception of TRISO-coated particle fuel now being tested for use in modular HTGRs (and potentially in FHRs), no irradiations, PIE, or safety testing of molten salt fuel is underway or planned in ART. Thermal neutron irradiation environments are available to support most expected testing for thermal MSR designs, but fast neutron irradiation support facilities are not available domestically and must be established to support fast MSR concepts.[m]	Medium	Legacy MSR-E fuel performance data exists, but is adequate for only preliminary safety analysis of fuel designs similar to the MSR-E; these data are less useful as a concept moves away from the historic MSR-E design. No domestic fast irradiation capability currently exists for a fast design like chloride salt. Fuel safety testing is likely a long lead time activity with a major influence on licensing timelines. Assuming resource constraints allow ART to support basic "generic" work plans that support multiple MSR concepts, this topic is a medium licensing priority pending further clarification on developmental emphasis.[n] ART MSR technology R&D will require careful planning to develop generic studies on fuel service conditions that benefit the entire technology class. The priority should be re-evaluated when the ART R&D approach is available concerning fuel type, available knowledge, and needed testing.
3	**Demonstrate fuel performance requirements are met for accident conditions using irradiated fuel at accident conditions and monitored fuel accident performance**								
3.a	HTGR	Demonstrate the scope of fuel performance testing for LBE accident conditions is adequate. Ensure conditions like reactivity excursion events, moisture-ingress events, and air-ingress events are adequately understood and factored into fuel performance requirements.[a]	High	Test results on this topic are important to the interface between LBE selection and associated accident analysis and consequence predictions. This, in turn, is essential to FQ and MST development.	Medium	AGR irradiations/PIE/safety testing were initially planned to provide data over a broad spectrum of fuel performance and fission product transport conditions within fuel particles, compacts, and graphite materials representative of fuel element blocks or pebbles. Additional data, coupled with in-reactor measurements (i.e., irradiation conditions and fission gas release-rate-to-birth-rate ratios) are necessary to definitively demonstrate compliance with established fuel performance requirements and support development and validation of safety computer codes.[b]	Moisture and air ingress events are considered in the context of fuel performance in the AGR test plan.[b] This topic will be characterized at the conclusion of AGR Test 5/6/7 PIE (scheduled for 2024). Existing reactivity excursion data are sufficient to support design decisions and initial licensing activities. No other R&D plans exist for further reactivity testing.[d]	Medium	Firm understanding of fuel performance during design basis accident conditions is critical to licensing success. Although the exact nature of LBEs remains to be confirmed, adequate test data currently likely exists or will be generated under the AGR program to address this concern. Licensing priority is set recognizing that R&D is underway to resolve the topic and scheduled for closure without impacting the licensing critical path. A process for selecting LBEs for licensing assessment remains to be established with NRC.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
3.b.	HTGR	Perform irradiation testing, PIE, and safety testing of qualification test fuel to demonstrate the reference fuel meets fuel performance requirements expected of HTGRs during accident conditions. Obtain data needed for fuel performance model validation.[b]	High	This activity provides fuel performance information and irradiated fuel samples for PIE and post-irradiation heating tests in sufficient quantity to validate fuel performance codes/models and demonstrate the capacity of the fuel to withstand expected conditions. The information is essential to support detailed plant design and licensing reviews.	Medium	When completed, AGR test regimes will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE in sufficient quantities to demonstrate compliance with statistical performance requirements under normal operating and anticipated LBE accident conditions.[b]	Fuel qualification testing and PIE/safety testing for accident conditions are scheduled in conjunction with AGR Tests 5/6/7. Irradiations are scheduled for completion by 2020 with PIE/safety testing finishing in 2024.[b] Data developed from these tests are expected to yield adequate data to support fuel performance model evaluations by both designers and NRC during regulatory safety review.	Medium	This activity has a very high level of licensing importance and a medium state of knowledge. Acquisition of additional data and information through AGR test #5/6/7 is essential. Licensing priority is set at “medium” to reflect planning anticipated completion of this R&D activity by 2024.
3.c	SFR	Ensure safety test information is available to ensure key fuel transient behaviors and parameters affecting modes of fuel failure are understood and factored into performance requirements. Perform additional testing to address important gaps. Ensure reproducible data supports transient fuel performance model validation needs and bound the physical phenomena that could degrade SFR fuel performance under off-normal conditions that include DBAs.	High	Knowledge about transient fuel behavior is needed to link LBE selection and analysis with fuel qualification and MST assessments. All physical phenomena that could significantly degrade SFR fuel and contribute to radiological source terms must be understood for the full spectrum of off-normal design conditions and enable predictions of fuel performance and consequences in the event of fuel failure. Fuel performance data and PIE of irradiated fuel samples are important to a licensing safety review and should be used to validate predictive fuel performance models.	Medium	Previously irradiated samples are now undergoing PIE and legacy data are available from safety testing and PIE done at EBR-II and FFTF. This information can be supplemented by information from incidents at SRE and Fermie 1 to demonstrate fuel performance during a range of postulated accident conditions.[f] Existing characterizations emphasize medium range burnup (<10 %) fuel, which are currently believed sufficient for initial SFR licensing under off-normal operational conditions; this assumption is contingent on the type of fuel used for the initial module and the data expectations and requirements of the regulator.[o] Experiments have been performed concerning fuel movement and transport during transient overpower conditions. Gaps for irradiated fuel beyond 10% and certain novel fuel designs (such as vented fuel) may require additional testing, however.[g]	Knowledge preservation regarding transient SFR fuel behavior is underway concerning EBR-II, FFTF, and TREAT safety testing databases. Qualification efforts are planned at ANL for key EBR-II and TREAT data.[i] Evaluation of data sufficiency for the full spectrum of design-specific LBEs have yet to be performed. There is significant reactor supplier interest in establishing additional fast reactor testing capabilities to reduce uncertainties and enhance efficiencies. Additional transient tests are being considered with FCRD Advanced Fuels, TerraPower, CEA Astrid, the Japan Atomic Energy Agency (JAEA), and through DOE’s GAIN initiative to address key areas of analysis that include fuel failure modes under loss of flow conditions.[p]	Medium	Ongoing data recovery and qualification efforts by DOE-NE ART are essential to SFR licensing. Data gaps may still exist relative to off-normal fuel performance but PIE on legacy samples and TREAT transient fuel testing capabilities should support a basic DBA and BDBE analysis. Additional fast irradiations are a long lead time activity with limited or unavailable supporting infrastructure at this time. Presuming the existing fuels knowledge base is adequate to characterize basic off-normal fuel performance and can be qualified, this activity is a moderate licensing concern for designs resembling EBR-II; other SFR designs may consider this a high priority in licensing success.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
3.d	MSR	Ensure MSR fuels irradiation safety testing and PIE information is available to support and verify fuel design safety goals. If data are needed outside the existing experience base, perform additional testing as required to address gaps across the DBE spectrum.	High	Fuel safety test information gathered through representative irradiations must be available during safety reviews to predict fuel performance over the full spectrum of off-normal conditions applicable to that design. It is understood that existing data may be applicable to some MSR fuel designs (i.e., TRISO-coated particle fuel used in FHRs), but all MSR fuels remain to be confirmed for use in a commercial reactor; supplemental investigative and confirmatory testing (including transient testing) will likely be needed in custom-designed testing loops and may be used to broaden design options, increase assurance that fuel design performance requirements are robustly met, and better understand relationships between fuel fabrication processes, fuel properties, and fuel performance under off-normal conditions that include design basis accidents.	Low	Traditional reactor fuel performance expectations do not necessarily hold true for MSRs. Many molten salt fuel designs now being proposed must rethink fundamentals of fuel qualification, performance, and safety not so much in terms of physical fission product release "barriers," but rather as control regimes based on parameters of buildup, cleanup, precipitation, retention, and release as a function of variations in coolant salt chemistry. Licensing focus remains on fuel system safety, but that safety considers factors like system lifecycles of radioactive material, physical retention and release mechanisms, potentially rapid changes in MSTs used in safety analysis, and how online refueling and/or salt processing might mitigate safety concerns and potential accident consequences differently.[l] Answers to such questions must be supported by performance demonstrations covering the spectrum of off-normal MSR operations and may extend into BDBEs. In general, these tests remain to be planned and performed at facilities that have yet to be identified.	With the partial exception of TRISO-coated particle fuel now being tested for use on modular HTGRs (and potentially of use in FHR concepts), no irradiations, PIE, or safety testing of molten salt fuel is underway or planned in ART. Thermal neutron environments are available to support most testing of thermal MSR designs, but fast neutron irradiation support facilities are not available domestically and must be established to support fast MSR concepts.[m]	Medium	Legacy MSR-E fuel performance data are adequate to initiate preliminary fuel design analysis similar to MSR-E; the data becomes less useful as concepts move away from historic postulated accident scenarios and depart from the MSR-E performance envelope. While TREAT can support transient tests, no domestic fast irradiation capability currently exists for fast designs using chloride salt. Fuel safety testing is a long lead time activity with major influence on licensing timelines. Assuming ART resource-constraints allow only "generic" R&D that helps all MSR concepts, the topic is a medium licensing concern pending definition of the ART technology R&D focus.[n] Once developmental emphasis is established and off-normal fuel service conditions are better defined, the priority should be re-evaluated in terms of available fuel knowledge and characterization needs.
4	**Establish and validate models for fuel performance and radionuclide transport in fuel**								
4.a	HTGR	Perform irradiations, PIE, and safety testing of qualification test TRISO fuel to support fission product transport code validation.[b]	High	A fission product code that is used in safety evaluations must be validated by appropriate data, reviewed and endorsed by the NRC before use in safety assessments. This activity is essential for development of a MST that can be accepted in licensing decisions.	Medium	A multi-monitored AGR test train will be used that includes fuel compacts seeded with fuel particles that are missing buffers. The fuel particles will then be subjected to different temperatures among various capsules. The test train will provide irradiated fuel performance data and irradiated fuel samples for PIE and safety testing to validate fission product transport codes.	Experiments associated with AGR Test #8 were initially planned for irradiation in the ATR flux trap housed in one test train or a "Large B" position.[d] Data from a test like this will be needed to validate the fission product transport code. However, performance of AGR #8 has been put on indefinite hold due to lack of relevant design-specific details and resource constraints.[b]	Medium	This activity is significant to licensing but has a medium level of knowledge. ATR # 8 results will be needed to address outstanding MST concerns. The test has not been scheduled due to insufficient support. Once a design becomes known to support AGR #8 planning, the test should be performed on a schedule coordinated with application development. The need for design information from applicants lowers the licensing priority to "medium."

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
4.b	HTGR	Resolve outstanding uncertainties concerning flux-accelerated diffusion of metallic fission products in TRISO fuel during irradiation.[a]	Low	This issue is not of significant regulatory importance. It is an issue about collecting confirmatory information about a topic DOE/INL believes to be already accurately characterized.	High	A central tenant of AGR irradiation, PIE, and safety testing is to obtain data on fission product transport through the fuel matrix and graphite with known sources of fission products resident in the fuel. This allows measurement and evaluation of the fission product gradient across the matrix and graphite surrounding the fuel through PIE.	Critical reviews and analysis of historic data on both for in-pile and out-of-pile fission product diffusion in TRISO-coated particle fuel is underway. For AGR tests, DOE/INL will use PIE to measure the release of fission products under irradiation, analyze the measurements to establish diffusion coefficients under irradiation, and compare the resulting diffusion coefficients to the historic values taken from IAEA-TECDOC-978.	Low	Further research in this area yields a low regulatory impact. The current state of technical knowledge is good and should suffice for licensing purposes.
4.c	HTGR	Confirm radionuclide transport assumptions for the compact-to-graphite gap of the prismatic fuel element.[a]	Low	For HTGR LBE transients, the effects of compact matrix and graphite sorptivity on metallic fission product transport across gaps are (conservatively) neglected. NRC staff view this approach as reasonable in the context of conservative consequence analysis.[a] This issue has insignificant regulatory impact.	High	Calculation of event-specific MSTs for the prismatic core design presumes the fuel compact-to-graphite gap to have no effect on the transport of gaseous fission products.	Further research on this issue was incorporated into AGR 3/4 fuel tests. Conclusion of those tests and PIE (scheduled for 2020) will fulfill remaining data needs.[d]	None	Because conservative presumptions are already made with respect to safety, further research on this topic does not incur any licensing concern.
4.d	HTGR	Develop transport models for all radiologically significant radionuclides in TRISO-coated particle fuels.[a]	High	Representative and robust capabilities to conservatively model and predict radionuclide transport from point of generation (within the fuel kernel) through all barriers to offsite receptors is a critical element in MST assessment and an essential component in independent safety reviews.	Medium	It is the position of DOE/INL that collection of data on all radionuclide species that could be of concern during MST calculation is unnecessary. Instead, DOE/INL has proposed to classify each radionuclide and species into one of nine representative radionuclide classes (established based on chemical and transport property similarity) and conduct subsequent analysis according to the class properties rather than a comprehensive species-specific analysis.	DOE/INL is working on developing experimental data sets for fission product transport of representative classes of radionuclides (e.g., Cs-137 for alkali metals, I-131 for halogens) and applying that to model all other radionuclides in that class.[b] The AGR test program has not scheduled extensive testing related to tritium transport, however.[d]	Medium	The data application approach being proposed appears sound with respect to licensing. However, NRC review and approval of the approach will be required of NRC. HTGR outlet temperatures at or below 750°C do not create major tritium transport concerns but a significant gap will be created if design outlet temperatures increase significantly beyond 750°C. Further interactions with NRC are needed once preliminary design is known and regulatory transport models are developed.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
4.e	HTGR	Develop data and predictive models for particle fuel performance during normal operations, heat-up events, and reactivity accidents. Include consideration of accidents with attack by oxidants and determine effects of air and moisture ingress on particle coatings and exposed fuel kernels.[a]	High	Accurate and valid models that predict TRISO coating degradation and failure mechanism under normal and off-normal conditions are essential to assess risks to public safety. Experimental data must adequately envelope all LBEs (including DBAs) that involve air or moisture ingress as may be present in the final design. Air and moisture (at a minimum) are known to potentially affect particle failure fractions and releases of iodine, metallic fission products, and fission gases.	Medium	DOE/INL uses the 1989 Goodin-Nabielek model for fuel performance. Understanding important material properties is necessary for accurate modeling under normal irradiation and accident conditions. However, the ability to obtain applicable data is limited by resources and (in some cases) by particle measurement science.[b] Fuel energy deposition and maximum fuel temperature for most limiting reactivity insertion accidents is low and depends on plant-specific design and analysis (which are yet to be established). DOE/INL R&D efforts have concluded that oxidant contaminants will encounter extensive reactive material before reaching fuel particles despite relatively rapid oxidant diffusion through matrix materials.	The AGR test program will develop fuel performance information of coated-particle fuel (UO_2 or UCO) that are first principle-based and include a prioritized list of material properties and constitutive relations needed for accurate modeling of coated-particle fuel under normal and off-normal conditions.[b] Design and analysis details remain to be established to determine whether fuel testing specific to HTGR reactivity excursions is necessary. Experimental measures of fuel element graphite oxidation and fuel element matrix during representative air and moisture ingress conditions are addressed in research plans for moisture and air ingress.[b]	Medium	Particle fuel performance models for normal and off-normal operations are a major regulatory concern. Much of the necessary support data is already available and basic model development is underway.[a,c] Subsequent research efforts may be needed as a result of forthcoming design-specific decisions, but such efforts are contingent upon choices made by the designer. Because needed R&D is actively in-progress, the licensing priority is established as “medium.”
4.f	SFR	Identify, describe, and confirm significant radionuclide transport phenomena and related assumptions for SFR fuel. Develop fuel behavior analytical models to predict margins to cladding failure and related contribution to source terms during postulated accidents.	High	A validated predictive capability for margin-to-cladding-failure assessment during postulated accidents is essential for a design safety review. The unique effects of fuel pin sorptivity and interaction with sodium coolant plays a potentially major role in fission product transport for SFRs and must be characterized and quantified with appropriate safety margin.	Medium	Development of fuel behavior models for a representative spectrum of LBEs (including accidents that could lead to fuel failure), is being pursued under DOE-NE’s ART program.[g,q] Experimental test results concerning radionuclide movement in fuel and transport data are available for validating models. Radionuclide release from metal fuel is well understood for cladding failure scenarios and low-burnup fuel melting. However, it may be found that a more mechanistic approach to modeling radionuclide release into sodium from molten metal fuel at high burnup is needed.[m]	Current ART efforts are focused on legacy data recovery/qualification and model tool development. An approach to address test data compliance with NQA-1 requirements has been developed and implemented.[i] A survey of existing research SFR code capabilities has been conducted and improvements/validation of a core analysis code has been started.[o,q] Once this code is updated/validated, regulatory endorsement of these tools can be sought.	High	Continued development of fuel behavior understanding and radionuclide transport models based on mechanistic approaches (and appropriate validation of these models using legacy or contemporary data) is a high priority essential to both reactor design and licensing processes.
4.g	MSR	Identify, describe, and confirm significant radionuclide transport phenomena and related assumptions for MSR fuel. Develop fuel behavior analytical models that predict margins to fuel failure in relation to source term contributions during postulated accidents.	High	A validated predictive capability for fuel failure and assessment of related consequences during postulated accidents is essential to completing a design safety review. Unique interactions between the fuel and specific molten salts that produce, transport, and release fission products must be thoroughly characterized and quantified (with appropriate safety margin) to assure public safety during all LBEs.	Low	MSR technology currently embraces both fast and thermal reactors with variants from chloride to fluoride salt and solid to liquid fuels. Modern modeling and simulation (M&S) tools, as well as validation data tailored to the subtype, are needed for licensing. Liquid fuel MSRs are unique due to convection of delayed neutron precursors and transit times through the core and the remainder of the primary loop. M&S tools are available but need to be adapted for MSR refined evaluations.	Simplified models that replicate MSR-E dynamics are being recaptured; initial planning has started to identify information gaps and additional needs.[s] Functional requirements for core M&S tools remain to be established and validation data generated for each specific design case. FPT depletions with continuous and batch feeds and removals remain to be characterized.	High	Foundational fuel performance and radionuclide transport analysis questions include the type and form of fuel, the employed salt, activation and fission product life cycles, system behaviors, and the role fuel plays in LBE scenarios. While validated fuel performance and FPT M&S tools are essential to licensing success. The low state of MSR MST maturity suggests a high licensing priority for development.[n]

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
5		**Develop fuel product and fuel fabrication process specifications**							
5.a	HTGR	Develop justifications for fuel design, fuel fabrication product specifications, and fuel fabrication process specifications that support TRISO-coated particle fuels and the HTGR safety case.[a]	High	Due to the role TRISO-coated particle fuel plays in the HTGR MST, establishing and meeting fuel specifications becomes a critical element for demonstrating the overall design satisfies top-level NRC regulatory requirements. These requirements are stated in terms of dose consequences for offsite residents, occupational exposures, siting, and safety goals and objective for doses below Environmental Protection Agency (EPA) protective action guidelines at the site boundary for all LBEs.	Medium	The AGR TRISO fuel qualification program includes attention to develop fuel product and process specifications for large-scale TRISO fuel fabrication processes. This will define the requirements the fuel must satisfy to ensure acceptable fuel performance in the HTGR core as it operates normally and under DBA conditions.[b]	Key elements of fuel design and manufacture remain to be finalized. However, the initial fuel specification has been established and will be validated in AGR tests 5/6. The procedures and specifications for manufacturing TRISO particles remain to be documented in a topical report for submission NRC.	Medium	Developing a TRISO-coated fuel specification is crucial to fuel manufacturing and plant licensing. A fuel specification is available to support AGR tests 5/6 (scheduled to start irradiations in 2017) to confirm integrity of overall TRISO fuel design. Although important, licensing priority is "medium" in recognition of planned R&D activity; failure to satisfactorily complete these tests will cause the topic to become a very high concern for near-term HTGR licensing success.
5.b	HTGR	Develop and demonstrate a fuel fabrication process that equals or exceeds fuel fabrication requirements as required by applicable source term calculations. The process is to include adequate margins of safety and address influencing factors, such as heavy metal contamination, as-manufactured fuel particle defect rates, and in-reactor fuel performance.[a]	High	HTGR nuclear safety is uniquely dependent on a highly reliable and predictable TRISO fuel fabrication process that consistently complies with established specifications. Demonstrations on high-reliability TRISO fuel fabrication processes are also used to verify final fuel performance acceptance in terms of fuel particle failure rate and fuel radionuclide transport characteristics during normal operations and off-normal conditions.	High	DOE/INL and its fuel supplier (BWXT) has developed a substantial body of technical information on the manufacture of TRISO-coated fuel that satisfies required TRISO-coated fuel particle failure rate specifications during normal operation and heat-up (simulated) accident conditions. Fuel coating process development has been accomplished in two phases: the first was conducted in a 2-inch diameter laboratory coater and the second scale-up to a 6-in. prototypic production-sized coater.[b] This process was proofed in AGR #1.	The TRISO fuel fabrication process and product specifications for fuel qualification tests are set and the fuel used for qualification testing has been fabricated. Challenges remain concerning how to "optimize" the fabrication process.[d]	Low	Essential research in this area is complete. Existing levels of knowledge concerning fuel fabrication capabilities are very good and can support a licensing topical report for submission to NRC for review and endorsement. Licensing priority is "low." A generic particle fuel "topical report" is recommended for submission to NRC at this time with current AGR information for regulatory review.
5.c	SFR	Develop fuel design, fabrication, and process specifications to reliably produce fuel with requisite levels of quality. Include adequate margins for safety and factors that affect in-core fuel performance.	Medium	Establishing appropriate SFR fuel design specifications is required as a part of overall fuel qualification efforts. Once fuel design and performance specifications are known for a given design, conformance to those fabrication process specifications need to be demonstrated during licensing.	High	EBR-II and FFTF fuel design and fabrication experience is available to support topical development but that information remains to be formally assessed and confirmed for usability, quality, and comprehensiveness relative to emerging commercial designs.[e,h] If suppliers use a new approach like vented fuel, new fuel characterization information and measurement methodologies are needed. For proposed designs that use advanced alloys for cladding materials (i.e., HT9M), additional design, fabrication, and process specifications are necessary.[k]	Criteria for assessing SFR fuel performance have been established using experiments and test results from EBR-II, FFTF, and the TREAT facility. Heritage test information is being recovered and qualified for licensing use under DOE-NE's ART program.[q] Gaps in data relative to individual applicant designs needs remain to be assessed; addressing critical data gaps that require a fast irradiation capability will be a long lead time R&D activity (if needed).[f]	Low	Qualifying existing fuel specification and fabrication records is essential to fuels development approaches. Should vendors depart from legacy data coverage to seek increased tolerance in fuel failures or vent fission products, early pre-licensing interactions with the NRC are necessary to establish supplemental research plans that support licensing. Until supplemental fuel development needs are identified, this topic is considered a low SFR licensing priority.

Table 1. (continued).

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5d	MSR	Develop fuel design, fabrication, and process specifications to reliably produce fuel with requisite levels of quality. Include adequate margins for safety and factors that affect in-core fuel performance.	Medium	Establishing appropriate MSR fuel design specifications is required of the design as a part of overall fuel qualification efforts. However, the precise role fuel fabrication processes play in maintaining overall MSR safety remains to be firmly established and will drive the regulatory significance of R&D on this topic. Once fuel design and performance specifications are known for a given design, conformance to those fabrication process specifications need to be demonstrated during licensing.	Low	A wide variety of fuel product concerns still require characterization and evaluation in the context of safety approaches employed by MSR designs (the diversity in MSR concepts currently detracts from making general conclusions in this area). The issue involves basic fuel usage considerations on heterogeneity and swelling, delayed neutron migration, presence or lack of cladding, moderators, interaction between fuel and coolant, xenon and samarium override, the effects of fuel makeup systems, and the role fuel plays in assuring safety.[j] Fluoride salt reactor concepts can take advantage of MSR-E work, but chloride salt concepts require more testing.	There is no work underway on this topic in ART.[k] It is difficult to focus R&D interest with so many competing and widely divergent fuel concepts. While early salt reactor experience was encouraging and could provide a basis to start development of a fuel product specification, fast chloride reactor concepts will be different and difficult to test due to lack of irradiation facilities.	Low	A general lack of confirmed understanding the role MSR fuels plays in safety and the accompanying performance that will be required of the product fuel makes this topic an uncertain licensing priority at this time. Thus, it is given a low priority pending definition of MSR "fuel qualification." This will likely increase as the general safety case approaches maturity.
6	**Conduct irradiation and accident proof testing of fuel representatively fabricated on production lines of fuel fabrication facility**								
6.a	HTGR	Conduct irradiation proof testing and post-irradiation heating of fuel produced in a full production TRISO particle fuel fabrication facility to demonstrate acceptable quality and performance of fuel.[a]	Low	Irradiation proof testing and post-irradiation heating tests of TRISO fuel will be necessary to demonstrate acceptable performance and qualify the production line fuel. These tests will be required for TRISO fuel to verify production line fuel performance equal to that demonstrated in prototype testing applications.	Medium	DOE/INL used mixed batches of fuel made on a single production-scale line for AGR tests. This will simulate the variability of fuel made on fuel fabrication facility lines for the prototype.[a]	This activity is not currently included in the AGR Fuel Program. Fuel proof tests rely on data generated by PIE and post-irradiation heating tests generated by in-core irradiations. Fuel vendors would perform necessary confirmatory tests of the precursor AGR test data. The proof tests are to be performed later by the commercial-scale fuel vendor and are not essential for initial plant R&D/licensing purposes.	None	Manufacturing tests are expected to be confirmatory of predecessor AGR test data. No ART R&D attention is necessary or currently directed towards this issue.[d] Because NRC staff identified this as a future agency concern, applicants must address this activity during the development of the license application.
6.b	SFR	Determine if production line proof tests of fabricated SFR fuel are necessary as a function of associated potential to contribute to MSTs and affect plant safety.	Low	Demonstrating fuel fabrication specifications for SFR fuel are met is a fuel vendor issue. The extent and surety of data needed to demonstrate compliance will be primarily driven by fuel performance expectations set by the design. NRC will review this information during licensing.	High	No SFR fuels are currently being made domestically. However, there is a historic record concerning fabricated fuel for EBR-II and FFTF. This record is anticipated to be adequate to support near-term SFR fuel fabrication needs in prototype testing.	No efforts are underway to conduct research in this area beyond the recovery, review, and qualification of relevant historical data.	Low	The need for additional research on the topic is not yet recognized as a priority for SFR fuels types. Substantial experience exists that should enable licensing of the initial plant.
6.c	MSR	Determine if production line proof tests of fabricated MSR fuel are necessary as a function of potential contributions to MSTs and plant safety.	Low	Demonstrating attainment of fuel fabrication specifications for MSR fuel will be a fuel supplier issue. The extent and surety of data needed to demonstrate compliance will be driven by fuel performance expectations set by the design. The NRC will review this information during licensing. Given the state of current understanding, regulatory importance is low at this time.	Low	No MSR fuels are being made domestically. Historic records concerning fabricated MSR-E fuel are available as a starting point for further development. Key definitions related to how MSR fuel relates to plant safety and the proof testing regime required of that fuel remain to be established. It is presumed that all MSR fuel forms will need testing in an irradiation environment representative of the particular MSR concept.	No ART R&D is currently underway on this topic.	None	A scarcity in basic understanding concerning the role MSR fuel plays in the plant safety basis, along with the wide variety of fuels and fuel forms being proposed for use, preclude meaningful assignment of a licensing priority for ART R&D at this time. This topic is not yet assigned a priority ranking.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
7	**Develop event-specific mechanistic source terms**								
7.a	HTGR	Calculate a MST for HTGRs using accepted LBEs that demonstrate compliance with 10 CFR 100 requirements and the safety expectations conveyed in the Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants.[a]	High	An MST that is plausible, conservative, and has an acceptable level of uncertainty when applied to bounding LBEs is critical to evaluations of safety and establishing the nominal size of the site boundary and emergency planning zone.	High	Modular HTGR precedents and the AGR fuel test program results provide a sound technical basis for MST development. NGNP regulatory white papers and pre-licensing public meetings with NRC have conveyed specific positions and approaches on how MST should be calculated and used in siting decisions. The staff generally found these proposals reasonable, but final regulatory acceptance remains to be confirmed through a licensing action.[a] A regulatory pathway for selecting LBEs remains to be established.	Additional data pertaining to MST development is being collected through AGR Tests 3/4.[d] When PIE is completed, the primary hurdle remaining in resolving this issue will be associated with pre-licensing regulatory interactions between NRC and applicants. Future applicants must develop a licensing plan, which resumes regulatory discussions with NRC staff early to finalize prior NGNP proposals.[u]	Low	Adequate technical information is or soon will be available through the AGR Fuel Test program to support MST refinement. Unaddressed technical gaps on this issue are not significant with respect to additional R&D needs at this time. Given the state of HTGR fuel technology development and the technical elements (i.e., MST definition for siting and emergency planning) that remain to be completed, this issue has become more of a regulatory concern rather than an R&D issue.
7.b	SFR	Develop a representative MST model for bounding SFR LBEs that can demonstrate compliance with 10 CFR 100 requirements and the safety expectations conveyed in the Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants.	High	A representative MST that characterizes radiological releases for all operational modes and postulated accidents is critical to assessing SFR plant safety and defining necessary site boundary and emergency planning zones. Without reliable source terms (that include appropriate margin), a plant regulatory safety analysis cannot be completed and a license will not be issued. Given the unique nature of elements contributing to the SFR MST, early interactions with NRC staff are necessary to ensure MSTs are developed adequately representative of the specific design being reviewed.	Medium	A technical basis built on historic EBR-II data, past experimentation and metal fuel accident information, was used to develop a trial MST for a generic metallic-fueled, pool-type SFR (the type is nearest deployment).[r,v,w] Design concepts that call for vented fuel or otherwise deviate significantly from the historic generic model may require new information that requires additional irradiation testing.	ANL has characterized the history and major (qualitative) components of a conceptual metallic fuel, pool-type SFR MST; prioritized recommendations for future SFR MST R&D have also been identified.[r,v] Some R&D into MST contributing elements (particularly those dealing with accident conditions) need to be defined and developed when design-specific LBE scenarios are understood from suppliers.[i]	Medium	A MST covering the LBE spectrum is essential to regulatory safety reviews. Extensive operational information and safety test data is available as are details from past accidents. Gaps in a conceptual metallic fuel, pool-type SFR MST have been identified and recommendations have been made concerning further R&D options.[r] Until additional design details become available to support MST refinement or call for additional testing to support new design features, the topic is assigned a medium licensing priority.
7.c	MSR	Develop a representative MST model for bounding MSR LBEs that can demonstrate compliance with 10 CFR 100 requirements and the safety expectations conveyed in the Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants.	High	A representative MST that characterizes radiological releases for all operational modes and postulated accident conditions is critical to assessing plant safety and defining necessary site boundary and emergency planning zones. Without reliable source terms (with appropriate margin), a plant regulatory safety analysis cannot be completed and a license will not be issued. Given the new and unique nature of MSR design elements that will contribute to a MST, early interactions with NRC staff are necessary to ensure MSTs are developed that adequately represent the specific design undergoing NRC review.	Low	A generalized conceptual MST supported by test data for a generic MSR design has not been developed but may be useful to the technology class as a whole.[n] The beginnings of a MST technical basis may exist in MSR-E information, but the variations in current fuel and reactor design approaches, along with the still tentative nature of the overall MSR safety basis, means crafting the "generic" model envelope will be iterative and end up as a useful starting point for applying further design-specific options. From this initial MST model, a supplier could further "down-select" and refine certain elements to focus on their specific proprietary safety basis.	Although discussions are underway, no work is underway in ART on MST development for MSRs.[n] Basic design information such as fuel and salt to be used, along with basic core design considerations and the mechanisms relied upon to function as a fission product release "barrier" in a molten salt environment, remain to be defined and research planned to quantify these parameters in order to enable MST development planning.	High	A bounding qualitative MST model is crucial to licensing. However, inadequate design uniformity and incomplete safety basis information precludes comprehensively addressing the issue for now. Establishing a conceptual (qualitative) MST is a logical precursor to further MST R&D.[n] While an initial (conceptual) MST may be built on the basis of MSR-E, the model must be provisioned to allow further proprietary refinements and deviations by vendors. Creating this conceptual MST model should be a high licensing priority.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
8	**Establish and validate models for radionuclide transport to the environment**								
8.a	HTGR	Determine radionuclide transport behavior in the HTGR primary circuit and reactor building. Characterize impacts associated with reactor building vent/filtration system on MSTs.[a]	High	Understanding phenomena that influence radionuclide release and the ability to predict transport behavior of radionuclides in the HTGR primary circuit and reactor building is essential to establishing an acceptable MST. Modeling capability of these behaviors is required for regulatory safety reviews to satisfy siting requirements and design safety goals.	Medium	Correlations for predicting radionuclide re-entrainment during primary circuit depressurization transients have large uncertainties and are inadequately validated to support conservative predictions. Historic HTGR topical data is not extensive and large scatter is observed. Preliminary studies have been conducted to assess design options for the reactor building and the advantages and disadvantages of each option.	The AGR Fuel Program does not plan to perform single effects tests in an out-of-pile helium loop to characterize fission product deposition on and re-entrainment from primary system surfaces (i.e., plate-out and liftoff) under normal and off-normal HTGR conditions. Additional design-specific information will be necessary from applicants to support future research planning.[d] When the test is started, resulting data should be generated to validate methods describing transport behavior of condensable radionuclides in the reactor building under wet and dry conditions.[b]	Medium	Understanding specific radionuclide transport behavior to the environment is critical for demonstrating radiological safety in plant design. Predictive modeling capabilities of these behaviors will be examined during regulatory safety review to satisfy siting criteria and confirm design goals are met. Design-specific information from a supplier is essential to support research planning, thus making the topic a medium priority with respect to the technology licensing timeline.
8.b	SFR	Determine radionuclide transport behavior in the SFR primary coolant system and containment structure to support MST predictions during postulated DBAs.[e]	High	Understanding radionuclide transport behavior in the primary circuit and the low-leakage containment building proposed for SFR use is critical to developing a comprehensive MST model that enables impact evaluations at offsite receptors.	Medium	There are significant uncertainties in the unique role liquid metallic sodium plays in radionuclide retention and transport, thereby suggesting testing will be needed to understand the phenomena that support analysis code development.[r] Expert elicitation also suggests that the effect sodium plays in radionuclide transport is not well characterized (especially during accident events) and is a topic for future R&D. Developing understanding and fission product transport modeling in the SFR primary coolant system and containment responses is ongoing under DOE-NE's ART program.[q]	Substantial information from experimentation and accidents offer basic insight on radionuclide releases from metallic SFR fuels (e.g., during pin breach and with fuel melting at low burnup) and subsequent transport in the primary system. Additional radionuclide release testing from high burnup molten metal fuel may be necessary to optimize applications.[f] Research using representative radionuclide tracers (e.g., radionuclide release from fuel debris into a quiescent sodium pool and radionuclide behavior in containment), could be conducted using currently available facilities.	Medium	Understanding radionuclide transport behavior in the primary SFR circuit and in the low-leakage containment building is needed to developing a comprehensive SFR MST model. Uncertainties do remain and additional research options have been identified to address the gap.[r] Recognizing existing information may be adequate for initial licensing and plans will be developed to address design specific gaps, the topic is assigned a medium regulatory concern.
8.c	MSR	Determine radionuclide transport behavior in the MSR primary system and containment structure that support MST predictions during postulated DBAs.	High	Understanding radionuclide transport behavior in the primary circuit and the outer containment system proposed for MSR use is critical to developing a comprehensive MST model that enables impact evaluations at offsite receptors.	Low	Radionuclide transport behaviors are significantly influenced by chemical functions unique to and may vary significantly between MSR subtypes and fuel used (especially fuels that lack cladding). Examples include different fuel salt effects and changing cross-section over time. Dynamic behavior differences must consider boundary layers, turbulence, recirculation, and other fluid phenomena, along with temperature and density variations.[j] These dissimilarities create challenges to establish a representative radionuclide transport model for MSR technology as a whole.	Benchmark salt chemistry studies and analysis tools (in-pile and out-of-pile) need to be planned and performed to collect data that support analysis.[j,n] Interface chemistries that influence radionuclide transport remain to be studied, defined, and understood at levels that support MST development. DBAs also still require definition.	High	The numerous and varying MSR concepts being proposed are an obstacle to ART R&D planning; a technology review is advised to identify high-value cross-cutting research benefits the entire MSR community. The basic nature of radionuclide transport behavior, release, and control will require R&D at facilities that may not now be available. Fission product behavior and release is a core licensing concern and because extensive information must be gathered concerning basic behaviors, R&D is a high licensing priority.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
9	**Demonstrate mechanistic source terms models in best estimate and conservative analyses of transients and accidents**								
9.a	HTGR	Establish an evaluation methodology that addresses HTGR MST uncertainties and determine associated comprehensiveness. Include a basis for the terms "best estimate" and "conservative."[a]	High	MST models must show reasonable degrees of comprehensiveness and certainty to justify using in-siting and design safety decisions. The NRC must review and endorse the proposed approaches when they are used for purposes of determining safety.	Medium	Approaches for accident consequence analysis rely on calculation of event-specific mechanistic building-release source terms and associated dose rates, which are based on current understanding of radionuclide generation and transport phenomena. A Monte Carlo uncertainty analysis is used but can address only parametric uncertainties. Clarification of "best estimate" and "conservative" is largely a regulatory concern outside the nominal domain of active ART R&D planning.	The AGR fuel qualification program will generate test data that establishes and confirms MSTs under normal and accident conditions as accurate to within prescribed limits.[b] However, it is also understood that likely applicants do not have a developed capability to quantitatively develop such a methodology.[d]	Medium	Development of a methodology for addressing MST uncertainty is critical to plant siting and for characterizing the safety design basis. The NRC must review and endorse this methodology once it is developed. INL has the data and capabilities to develop the methodology, but the activity is not currently within the work scope. It is a medium licensing concern and an item to be addressed by applicants.
9.b	HTGR	Develop MSTs for specific HTGR LBE categories.[a]	High	Developing and using a MST is directly related to the LBE categories proposed by the applicant for use and accepted by NRC. The issue is critical to design basis evaluations of safety and siting acceptability analyses. Although extensive pre-licensing interactions have occurred with NRC staff concerning MST development approaches, similar interactions regarding LBE category development under the NGNP project proved inconclusive. Additional effort to develop this topic is currently underway within NRC.	Medium	Upon completion of currently planned AGR tests, the major technology elements of the HTGR MSTs addressable by R&D will have been characterized. Monte Carlo methods can be used to determine the overall effect of uncertainties on resulting source terms (including the fuel failure fractions and fuel radionuclide releases) and off-site consequences. These results can then be linked to formulate consequence distributions to provide a basis for judging acceptability and safety margins for a range of requirements.	DOE/INL will continue to develop source terms based on models already proposed to the NRC. The most important HTGR barrier to fission product release (i.e., coated fuel particles) will be modeled on a statistical basis to account for uncertainties about a mean in particle failure probability. However, linking source terms to specific LBE categories that represent the plant design basis requires specific design information from applicant(s) currently not available. An industry-led team is currently working with the NRC to establish a technology-inclusive approach to LBE selection (see Table 7 for further discussion).	Medium	Development of MST is essential and its application is a function of LBE category selection. Extensive pre-licensing interactions have occurred with NRC staff concerning MST development approaches for HTGRs and found to be reasonable. Selection of LBE categories remains a source of uncertainty and requires applicant involvement. Interaction with NRC staff is underway concerning a possible approach to LBE selection; additional interactions should be initiated by applicant on this topic during pre-licensing.
9.c	HTGR	Obtain peer review of MSTs.[a]	Low	Peer review is a standard component of the PRA development process and expected to be documented during NRC reviews.	High	Peer review of a PRA is a standard approach in the nuclear industry. The process is well understood and currently available for use.	PRA elements of MST development (e.g., LBE selection) will be peer reviewed, including source term calculations.	None	Peer review processes are not a significant ART research concern or a typical concern for licensing.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
10	**Develop prototypic pre-operational/operational programs to supplement/verify technical bases for fuel qualification and mechanistic source terms**								
10.a	HTGR	Evaluate the application of the NRC prototype provisions regulations to facilitate initial plant licensing.[a] This may include use of the prototype provision to verify and supplement the plant technical basis for items such as fuel qualification, fuel service conditions, fuel performance, and MST.	High	Potential undetected anomalous or off-normal operating conditions may require additional safety considerations when establishing initial plant operating limits. The presence of safety-related unknowns is a factor of on-going concern for NRC in both long-term and immediate pre-accident operating histories that are commonly used in a safety analysis. A safety analysis conclusion may require supplemental confirmations through applied prototype tests, surveillances, monitoring, and inspections.	Medium	The purpose of prototype-specific design development programs is to verify that the initial and subsequent operating conditions and performance elements (e.g., fuel performance) as developed based on research-level results are consistent with those predicted and considered in the technical basis for scaled-up plants. For a technology like HTGRs, a prior operating history does exist which can supplant the need for prototype-level demonstrations of safety.	No ART research is currently planned in this area. If prototype demonstration(s) of individual design element(s) is required by the NRC as a consequence of the independent safety review, DOE/INL can advise how design features, testing, and surveillance programs can be crafted in the initial plant specific to the necessary demonstrations. This input can assist future applicants in supplementing the developmental technical basis beyond that now being established.	Medium	Understanding requirements and resolving associated prototype plant issues that may arise during pre-licensing interactions will require involvement of designers and applicants. If prototype provisions are employed as a HTGR licensing option, interaction with the NRC should be initiated by the applicant and can be expected to yield additional licensing conditions for the initial facility. This topic is a significant licensing concern, but priority is reduced because the issue cannot be adequately addressed until a detailed licensing plan is developed for the first HTGR plant.
10.b	HTGR	Identify remaining challenges and the potential need for physical verification of normal fuel operating conditions in HTGR reactor cores.[a]	Medium	Accident source terms in modular HTGRs respond to core operating conditions. Inherent technical challenges in monitoring HTGR core internals during normal operating conditions makes measurements difficult to perform. Should in-core measurements be requested by the NRC to confirm safety analysis parameters and make it a condition of initial module licensing, instrumentation deployment and reliability during use may be a significant technical challenge.	Medium	Multiple factors contribute to difficulties in predicting normal operating conditions in prismatic-block and pebble-bed HTGR cores. In both pebble-bed and prismatic reactors, typical operating temperatures are too high for most thermocouples. Additional thermocouple development could overcome this limitation. However, for pebble-bed reactors, instruments cannot readily be inserted into the core. Melt-wire pebbles could be dropped into the core to obtain data on peak coolant temperatures from which local fuel temperatures can be calculated. It is difficult to precisely place and track these pebbles, however, which underscore the uncertainties inherent to the process.	DOE/INL can aid in developing approaches and plans for performing in-core measurements in the HTGR demonstration plant to verify normal core operating conditions and demonstrate adequate detection of operating condition anomalies. No research is currently underway in high-temperature thermocouple design, however. Getting precise in-core temperature profiles will be difficult but combinations of some measurements with new thermocouples and better core simulation capabilities could bound uncertainties in the core temperature profile.	Low	Current thermocouple technology does not fully enable HTGR in-core temperature monitoring. However, it is possible the NRC will require in-core monitoring for the first module. Additional R&D could develop the needed technology if required; development will require applicant involvement and commitment. This issue is a potentially serious one in licensing success, but actual need remains to be confirmed in conjunction with COL application development. The topic is a low-licensing priority at this time.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
10.c	SFR	Evaluate regulatory prototype provisions to facilitate licensing of initial SFR unit. Consider needs to verify and supplement the technical basis for fuel qualification, MST development, establishing fuel service conditions, and confirming projected fuel performance. An initial prototype plant may be needed to address major design and operational concerns.	High	Without adequate support data and information about plant safety and margin, commercial SFR licensing may require a prototype deployment to assess uncertainties and refine operating limits. Prototype operations require larger safety margins and additional measurement, testing, surveillance, monitoring, and inspection programs. There is little precedent in operating a plant under prototype NRC regulations, thus making the licensing approach complex and uncertain when used on a demonstration plant-scale basis; early interaction with NRC staff is mandatory when considering the prototype process.	Medium	Due to historic limitations in SFR licensing experience and a scale-supporting infrastructure capable of testing areas like fast reactor fuel qualification, a prototype approach to SFR deployment may be mandated for initial plant licensing.[h] A fast reactor testing platform will be needed to qualify fuels.	Full-scale prototype plant operations are currently presumed unnecessary for SFR designs that resemble EBR-II; ART R&D is not geared to support a major SFR prototype demonstration plant deployment. Designs that exceed the boundaries of historic EBR-II design and operations will likely be found to need a prototype plant for initial licensing or rely on key prototype monitoring programs to collect essential safety data necessary for later reactor licensing.[z]	Medium	Pending confirmation on initial unit design, it is presumed a full-scale prototype plant option will not be necessary to license the first SFR. However, certain activities such as fuel qualification will need access to a prototypical-scale fast reactor test platform of the regulatory review dictates it is needed. This activity is given a medium level of regulatory concern at this time and may increase once actual prototype research needs are better defined.
10.d	MSR	Evaluate regulatory prototype provisions to facilitate licensing of the initial MSR unit. Consider needs to verify and supplement the technical basis for fuel qualification, MST development, establishing fuel service conditions, and confirming projected fuel performance. An initial prototype plant may be needed to address major design and operational concerns.	High	Insufficient support data and information about plant safety and margins will require MSR licensing to rely on a prototype-scale deployment to assess uncertainties and refine operating limits. Prototype operations require larger safety margins and additional measurement, testing, surveillance, monitoring, and inspection programs than otherwise required. There is little precedent for operating a plant under prototype NRC regulations, thus making the approach complex and uncertain when used on a plant-scale basis; early interaction with NRC staff is mandatory when considering the prototype process.	Low	Due to the very limited range in MSR operating experience, a prototype reactor is very likely needed. ART has identified an engineering demonstration or commercial prototype as required, but not specified the MSR type to be built.[l] ART is considering plans for several developmental options in non-proprietary design features that can aid deployment schedules in the 2030 to 2035 timeframe. Some prospective MSR suppliers believe historic MSR-E experiences will suffice for their salt-fueled, thermal spectrum “demonstration” reactor, but this remains to be confirmed with NRC and would be inadequate for fast chloride salt reactors.[l]	Roadmaps leading to higher MSR readiness levels are anticipated to be built from the ART Advanced Demonstration and Test Reactor (ADTR) study.[l] A decision to proceed with ADTR facility development has not yet been made.	Low	Technology tests and demonstrations should occur early enough to influence subsequent stages of commercial prototype designs. A MSR technology down-select is needed to ascertain the nature and need for prototype regulation application in MSR design. This R&D topic is given a low licensing concern pending greater clarity on design choice and confirming actual regulatory need for the technological application. Once R&D focus is outlined, this topic will likely increase in licensing priority.

Table 1. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification

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b. INL, "Technical Program Plan for INL Advanced Reactor Technologies Technology Development Office/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev 6, June 20, 2017.
c. INL, Personal communication with H. Gougar, June 28, 2017.
d. INL, Personal communication with D. Petti, February 10, 2015.
e. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2," SAND2012-4260 & SAND2012-4259, May 2012.
f. ANL, Personal communication with D. Grabaskis, May 3, 2017.
g. ANL, Personal communication with T. Sofu, March 20, 2015.
h. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014.
i. ANL, "Quality Assurance Program Plan for SFR Metallic Fuel Data Qualification," ANL/NE-16/17, Rev 0, July 5, 2017.
j. Southern Research, Personal communication with Lance Kim, May 3, 2017.
k. INL, Personal communication with J. Carmack, May 3, 2017.
l. Qualls, A.L, and Hale, R.L., "MSR Technology Roadmap," DRAFT, ORNL/TM-2017/199, Oak Ridge National Laboratory, May 2017.
m. ORNL, Personal communication with J. McDuffee, May 3, 2017.
n. ORNL, Personal communication with L. Qualls, July 17, 2017.
o. ANL, "Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors," ANL-SMR-9, May 31, 2014.
p. INL, "FY2018 Integrated Strategic Transient Experiment Plan (ISTEP)," PLN-5318, February 15, 2017.
q. ANL, "Status of SFR Codes and Methods QA Implementation," ANL-ART-83, January 31, 2017.
r. ANL, "Regulatory Technology Development Plan, Sodium Fast Reactor Mechanistic Source Term – Trial Calculation," ANL-ART-49, Vols 1&2, October 2016.
s. ORNL, Personal communication with B. Ade, January 24, 2017.
t. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel," ANL-ARC-288, June 4, 2014.
u. INL, "NRC Licensing Status Summary Report for NGNP," INL/EXT-13-28205, Rev 1, November 2014.
v. ANL, "Regulatory Technology Development Plan Sodium Fast Reactor, Mechanistic Source Term," ANL-ART-3, February 28, 2015.
w. ANL, "Regulatory Technology Development Plan, Sodium Fast Reactor Mechanistic Source Term – Metal Fuel Radionuclide Release," ANL-ART-38, February 2016.
x. ORNL, Personal communication with G. Flannigan, January 8, 2014.
y. INL, Personal communication with H. Gougar, February 13, 2015.
z. ANL, "Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors," ANL/ART-88, Rev 0, April 20, 2017.

Description

Analytical Codes and Methods:

Developing and V&V computer models and analytical tools optimized to the appropriate non-LWR applications are essential to a safety analysis. Analysis techniques demonstrated to be acceptably reliable must be available to support system analysis and predictions of important phenomena, many of which may be unique to a particular design and/or safety approach. Often, these phenomena are initially identified through expert panel elicitation. Once a candidate's parameters are identified, a R&D strategy is applied that typically emphasizes: 1) identification of computer codes and support information/data needed to support both reactor design and NRC staff safety review of that design; 2) evaluation of existing computer codes and support information to identify gaps in both existing analytical capabilities and support information/data; and 3) interaction with domestic and international organizations that work to identify opportunities to collaborate in closing gaps. ART programs that work to close deficiencies in analytical codes and methods should be mindful about the need to maintain NRC technical review independence and that analysis codes created to support design maturation may also be later used by the NRC during an independent safety review. Assuring regulatory independence can be maintained during code development by assisting NRC staff in developing internal expertise in code use, phenomenological modeling, numerical schemes employed, and by adhering to objective, rigorous, and highly documented V&V processes.

Table 2. ART research regarding analytical codes and methods.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	**Define calculational envelope required to analyze reactor systems**								
1.a	HTGR	Identify the HTGR nuclear safety and performance envelope in terms of degrees of uncertainty regarding phenomena behaviors and the ability to predictively model.[a] Define scenarios required for licensing review/approval, perform scaled thermal fluid experiments, and identify key phenomena and figures-of-merit for important scenarios.[e]	High	Characterization of plant performance parameters that influence safety are an essential input to the regulatory safety analysis component of licensing. Comprehensive and objective data must be provided that support a comprehensive analysis, along with associated uncertainties that accompany the characterizations.	Medium	Challenges remain in the ability to model key phenomena that may influence safety. Phenomena modeling can be improved to quantify effects on core safety and performance. Thermal fluid phenomena still inadequately characterized include: air ingress after pipe break and blowdown; steam ingress after steam generator tube rupture; performance of passive vessel cooling system (especially water-based); heat transfer between blocks and across the core–reflector interface in pebble bed reactors (i.e., core heat transfer); extent of bypass flow between graphite blocks and its evolution with burnup; gravity-driven circulation of coolant plumes in the core after a loss of forced cooling and their effect upon the vessel upper head and control rod guide tubes (plenum-to-plenum heat transfer); magnitude of hot-streaking in the lower plenum; and subsequent propagation into the outlet duct.[k]	Major scenarios for HTGR safety analyses have been identified, key phenomena and figures-of-merit have been documented, and a model validation matrix formulated. Related testing is underway at Oregon State University's (OSU's) High Temperature Test Facility (HTTF) to address air ingress, ANL's Natural Convection Shutdown Heat Removal Test Facility (NSTF) to address vessel cooling performance, and JAEA's High-Temperature Test Reactor (HTTR) facility, which can provide physics data (e.g., rod worth, reactivity coefficients), pressure loss transient data, and performance of vessel cooling system data.[k] Model development, benchmarking, and uncertainty analysis of coupled neutronic/thermal fluid simulators will establish and characterize uncertainties in baseline core modeling capability. Benchmarking projects are continuing at several universities (under the Nuclear Energy University Program [NEUP]) concerning bypass flow, air ingress, and core heat transfer studies.	Medium	Key research is underway on this topic. Licensing priority recognizes enabling work is planned and underway that includes completion of test plans for HTTF, vessel cooling studies at NSTF (water-cooled studies), and plenum-to-plenum heat transfer studies. Other priorities include bypass flow studies, air/water ingress; coupled core and uncertainty analysis benchmarks; and computational fluid dynamics (CFD) simulations of core fluid and heat transfer phenomena to quantify potential errors in system/integral analyses.[k]

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.b	SFR	Identify SFR nuclear safety and the performance envelope in terms of involved phenomena, degrees of associated uncertainty, and ability to predictively model. Define operational scenarios that facilitate license review and identify key phenomena and figures-of-merit in importance scenarios of interest.	High	Defining the plant safety performance envelope is essential to regulatory review. In addition to design information, this involves data on reactor core physics, primary and intermediate heat transport system thermal-fluids, safety metrics, physical processes during normal and off-normal/accident conditions, and the capabilities/limitations of analytical models. Information must be tailored to address specific design features and the approaches affecting reactor safety.	Medium	Although many key parameters concerning a SFR design can be identified and quantified from historic EBR-II and FFTF experiences, gaps in understanding certain phenomena may exist or are associated with undesirably large uncertainties, particularly for design features that depart from experience base.[e]	Recovery of historic EBR-II and FFTF operational performance and test information, along with TREAT safety test data needed to analyze reactor system operation, is underway under DOE-NE's ART program.[f] The EBR-II records include test protocols and quality record recovery; these data will be placed into a searchable archive database and include component reliability information. FFTF recovery continues, but data handling and qualification measures have yet to be implemented.	Medium	While extensive DOE facility performance histories are available, data quality and completeness must be confirmed against contemporary designs and modern quality requirements. This is a medium licensing priority due to high regulatory importance and good state of existing knowledge; the priority is conditioned by presumptions that designs will remain within bounds of existing data.
1.c	MSR	Identify MSR nuclear safety and the performance envelope in terms of involved phenomena, degrees of associated uncertainty, and ability to predictively model the reactor system. Define operational scenarios that facilitate a license review and identify key phenomena and figures-of-merit in importance scenarios of interest.	High	Defining the plant safety performance envelope is essential to a regulatory review. In addition to design information, this involves data on reactor core physics, primary and intermediate heat transport system thermal-fluids, safety metrics, physical processes during normal and off-normal/accident conditions, and the capabilities/limitations of analytical models. Information must be tailored to specific design features and the safety approach used.	Low	The unique MSR design safety approach precludes use of most established calculational reactor system envelope analysis. Information to address this concern include multi-physics and multi-scale modeling, fundamental thermo-hydraulic, thermo-physical, and thermo-chemical characterizations of molten salts bearing actinides and fission products, reactor physics data on cross-section measurements, and code benchmarks.[g]	Multi-physics and multi-scale analysis tool packages remain to be defined, developed, and applied to benchmark systems that influence safety. These tasks could be performed at DOE national laboratories or at universities through NEUP once the envelope is defined. A flowing molten salt loop may be needed in a test reactor platform to validate molten salt property knowledge and reduce associated uncertanties.[g]	Medium	While extensive R&D is needed to address this topic, the ART regulatory priority will be meaningful after a suite of reactor system performance requirements focused on "generic" MSR attributes are established.[h] Licensing priority is medium as precursor fuel performance and qualification definitions remain to be established to guide envelope development.

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2	**Define evaluation models capable of an analysis across calculational envelope**								
2.a	HTGR	Identify and develop core and plant simulation tools of appropriate fidelity for modeling scenarios and phenomena important to HTGR safety that display large uncertainties or complex neutronic, thermal-hydraulic, and material interaction. Capabilities to perform 3-D simulations of core burnup and transients in HTGR reactors may be necessary.	High	Power reactors are licensed after compliance safety limits are established. Some limits are easily identified and implemented while others require complex models to establish and evaluate. Such modeling typically relies upon complex mathematical representations of the system. Many different models can be combined into a common computer code that represents major system phenomena. Complex codes used for regulatory safety analysis must undergo detailed confirmatory assessments to demonstrate they are appropriate and reliable for the proposed application.[i]	High	HTGR fuel design and related specifications historically assumed margin factors for core fission gas and metallic fission product release. These assumptions enabled approximations of fundamental design physics at levels that make it difficult to license a plant today. LWR analysis codes also missed features addressing layers of fuel and core heterogeneity and radiant heat transfer between pebbles and blocks.[j] Predictive model development and validation was needed to resolve such issues.[a] Neutronic phenomenon to be fully characterized in analysis tools include: physics of neutron scattering by graphite, elastic scattering in heavy metals; radiation damage effects on thermal properties of graphite; shutdown control rod voids (prismatic core); non-axial pebble flow and broken pebbles in the discharge cones (pebble bed core); and the extent of non-local fission energy deposition. While R&D is underway to address these and other similar issues, an inability to model these phenomena is not seen as a major barrier to licensing; lack of knowledge about thermal fluid behavior is more significant.[i]	Adequacy of early assumptions fission product release depends on outcome of the AGR fuel fission product transport data tests, AGR fuel qualification tests, and AGR fuel fission product transport code validation tests. The ART Methods R&D program is geared towards refining and using established existing software tools unless the capabilities of those tools are shown inadequate for HTGR licensing.[a] HTGR models using RELAP5-3D code are being validated using data from HTTF at OSU and elsewhere experiments as needed.[j] AGR test program data will significantly reduce uncertainties in modeling fission product transport through fuel compacts, blocks, and pebbles. Additional tool development coupling heat transport with fission product transport will enable better estimation of integrated fission product releases during steady state and transient operations. Improved phenomena modeling ability allows identification of further experimentation needs and quantifies source term sensitivity to various factors.[i]	High	HTGR fuel performance specification and analysis capabilities are key licensing concerns. Significant topical information is currently available and AGR test plan completion will extend the knowledge base to levels that generically support licensing. High fidelity multi-physics tools such as BISON, NEK5000, MAMMOTH, SCALE, PRONGHORN, and RELAP7 show great potential in CFD analysis and are subject to significant DOE-funded R&D.[j] ART VHTR methods R&D planned through 2021 are expected to fill most modeling capability gaps.[k] The critical nature of this work keeps activity completion a high licensing priority.
2.b	SFR	Develop and validate an analysis code system that is regulatory acceptable for primary and intermediate heat-transport system modeling and safety analysis tool.[g] Maintain the code system with a V&V test matrix and detailed documentation for V&V outcomes and detailed code descriptions to facilitate use. Improve modeling capabilities to include interfaces for high-fidelity multi-physics methods that reduce uncertainties in modeling integrated neutronic, thermal, hydraulic, structural phenomena, and characterization of processes that could contribute to MSTs.[d]	High	Power reactors are licensed after showing compliance with safety limits, which are sometimes established using complex model evaluations. Codes intended for regulatory use undergo a rigorous assessment to demonstrate appropriate fidelity and reliability for the application.[f] Properly characterizing important elements such as complex thermal mixing and changes in boundary conditions that could disrupt system performance may require high-fidelity CFD tools. Modeling processes that likely contribute to SFR MSTs are also of critical licensing concern.	Medium	While SFR safety analysis codes already exist, these are primarily R&D tools yet to undergo a rigorous and formal V&V process that support use during regulatory safety reviews. While the roots of these specialized tools are decades old, the codes have undergone periodic updates (for R&D purposes) by users both foreign and domestic. Because the codes have not yet been employed in a regulatory environment, they lack quality assurance demonstration and configuration controls required to support licensing.[n] A review has been completed that expressly identified the requirements and options available to code developers and users to bring the R&D analysis codes up to regulatory acceptance standards.[o]	The severe accident analysis system code SAS4A/SASSYS-1 is a major legacy research tool identified by DOE/ART as appropriate for multiple SFR licensing safety analysis scenarios. This code is being modernized, verified, and maintained under configuration management that supports eventual review by NRC staff and regulatory acceptance.[n] Existing data will be used to support the code needs to be assessed and confirmed in terms of its adequacy for a full event spectrum validation.[d] Other codes, including those from foreign sources that may be considered for use by some applicants as a safety analysis resource, also require detailed qualification and review prior to regulatory use.	High	Efforts were initiated within ART over the last two years to develop a SFR safety analysis tool (e.g., SAS4A/SASSYS-1); considerable modernization and V&V work remains to be done to bring this work to levels meeting regulatory expectations. Furthermore, the code contains uncertainties in relation to modern plant safety conditions, which are still being defined by designers. Until the code is reviewed by NRC staff for licensing use (and research planned to address gaps pertaining to that acceptance), the activity is considered a high regulatory R&D priority.

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.c	SFR	Develop and validate a regulatory-acceptable SFR fuel performance code. Maintain the code by preparing V&V test matrices and detailed documentation to facilitate regulatory reviews.[e]	High	Qualification of SFR fuel design and performance analysis tools (such as LIFE-METAL) are essential to completing a license safety review and enable a broad understanding of MSTs.	Medium	LIFE-METAL is an established R&D fuel performance code and a likely candidate for regulatory use in SFR safety analysis. The underlying validation database and documentation related to the model needs to be updated.[e] Work has been started to qualify legacy fuels data that can be used to support this validation.[f] NRC staff has not yet reviewed this or similar codes for purposes of regulatory acceptance.	Validation of the LIFE-METAL code requires completion and qualification of the EBR-II fuels irradiation and physics analysis databases, which are currently being developed under DOE-NE's ART program.[e,f] Development and maintenance of LIFE-METAL code is inadequately supported at this time.[p]	High	Validated fuel performance analysis codes are essential for licensing success. SFR code maturation hinges on recovery and qualification of heritage EBR-II and FFTF fuels irradiation experimental data, which is starting at ANL. Until a research plan is established to develop a qualified fuel code, the topic is a high licensing concern.
2.d	SFR	Update the MELCOR code with a CONTAIN-LMR module to cover phenomena related to sodium pool and spray fires and sodium-concrete interactions.[e,m]	High	The NRC relies upon a suite of analysis codes to support LWR licensing decisions. The confirmatory severe accident analysis code is MELCOR. Integration of SFR containment design analysis capabilities (CONTAIN-LMR) into MELCOR by incrementally adding to its radionuclide tracking capabilities is important to support subsequent regulatory MST evaluations.	High	MELCOR is a well-established regulatory LWR analysis code under formal configuration control. Integration of the CONTAIN-LMR module, which is not currently supported in the U.S., in MELCOR would create a well-maintained and accepted capability for radionuclide tracking, structure performance, and containment response analyses.[e] This "combined use" option is recognized by the NRC as a viable constrained code development environment resource by eliminating the need for separate primary analysis codes from confirmatory safety analysis code.[q]	The sodium-fire and sodium-concrete interaction analysis capabilities of CONTAIN-LMR are being modified for integration into MELCOR. MELCOR is an NRC code and adjustment will require involvement and approval of NRC staff and the NRC code configuration control authority.	Medium	Although updating MELCOR with CONTAIN-LMR capabilities is important to future licensing success, it is an adjunct to developing other essential SFR specific safety codes (e.g., Items 2b and 2c above). The R&D topic is given a medium licensing priority at this time.
2.e	MSR	Identify, develop, and validate core and plant simulation tools of appropriate fidelity for modeling scenarios and phenomena important to MSR safety. Characterize factors and uncertainties contributing to complex neutronic, thermal-hydraulic, and material interaction.	High	Power reactors are licensed after appropriate compliance safety limits are established. Some limits require complex models to establish and evaluate. Such modeling typically relies upon complex mathematical representations of the system. Different models can be combined into a common computer code to represent major system phenomena. Complex codes used for regulatory safety analysis must undergo detailed confirmatory assessments to demonstrate they are appropriate and reliable for the proposed application.	Low	Modern M&S tools exist to address regulatory requirements, but have not been amended to address MSR applications. Data are needed to begin optimizations and validations. Necessary capabilities include integral benchmarks for reactor physics, thermal hydraulics material properties and response models, coolant-fuel-structure chemistry/corrosion, and convection of delayed neutron precursors and transit of fuel through the core that is unique for molten salt-fueled reactors.[r]	There is no MSR-specific R&D currently underway within ART on this topic other than recovery of simple legacy code work derived from MSR-E experience.[h] Functional requirements for MSR core and plant simulation tools remain to be established. Suites of tools and quality data inputs must be developed for validation. Models must then be applied to specific design cases, which currently vary widely across the MSR design class.	Medium	Development of validated core and plant simulation tools to assess safety-related phenomena is essential for licensing. Different M&S analysis needs between MSR types (e.g., solid fuel systems vs liquid fuel systems) can be significant and suggests "generic" study sets are needed to guide ART R&D planning.[h] The topic is assigned a medium priority until detailed M&S tool performance requirements are defined.

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.f	HTGR SFR MSR	Develop and V&V new seismic analysis methods and integrated predictive models for seismic, structural, and plant systems analysis. Broaden applicability of existing seismic soil-structure interaction (SSI) computer codes to include deeply embedded or buried structures and SSCs where seismic isolation (SI) technology is used. Modify computer codes, such as MASTODON, to address the design and licensing need.[i,s]	High	No nonlinear soil-structure-interaction (NLSSI) programs, which are required for analysis of seismically isolated facilities, have undergone a regulatory V&V process. Some advanced reactor designs (e.g., HTGRs) call for reactor and steam generator systems to be built partially or completely below grade using deep embedments and will likely need SI features. Analysis of possible seismic events requires SSI. The seismic effects for these structures must be evaluated for predicted response. This assessment capability must be V&V with rigor to meet regulatory analysis requirements.[i]	Medium	Current SSI computer codes are based on past LWR designs where structure foundations are near ground surface. Developmental research for seismic analysis tools that evaluate reactor responses for deeply embedded or SI SSC remains to be initiated.[i] Additional development of the supporting database will likely be necessary.[t]	Seismic effect knowledge on key reactor performance attributes (e.g., coolant movement into or out of an assembly, core assembly distortions) in connection with subsurface embedment is insufficient for successful HTGR licensing using SI technology.[u] R&D is necessary for advanced reactor designs that employ a deep embedment or utilize SI equipment to assure safety. While initial work on the topic is underway at INL for LWR-derived SMR designs that employ an embedment, no research tuned specifically to SI and associated non-LWR design needs is underway. This work is estimated to take approximately five years and should be available in time to perform calculations in a relevant license application.[s]	Medium	Seismic analysis methodologies for deep embedments have not been identified or reviewed by NRC for use and may arise to intrude on the licensing critical path. Seismic isolation features used to enhance safety in a non-LWR design are not yet planned for development. Because R&D delays are not seen as impacting the critical licensing timeline yet, a medium priority is assigned. However, the activity will likely become a higher licensing priority once an applicant declares intent to license using SI within five years.
3	**Identify data or perform thermal fluid experiments to generate comprehensive database for validating design safety evaluation models**								
3.a	HTGR	Complete validation matrices for required analytical models. HTGR-related data used in model validation should address core physics, air/water ingress phenomena, bypass and lower plenum flow, core and plenum-to-plenum heat transfer, and seismic-induced geometry distortions, and other similar elements.[b] Design and run experiments (using acceptable scaling practices) where existing data is inadequate for computational dynamics and validation purposes.[a]	High	Developing, refining, and V&V of analytical models are critical regulatory safety analysis concerns. Data used to support models must be of high quality (i.e., meeting NRC quality assurance standards), complete, and address safety margins adequately. Data that support these models are subject to review and acceptance by NRC before those tools are used in licensing-related assessments.	Medium	Scenarios required for the HTGR analysis have been identified. Development and V&V of thermal, neutronic, and fluid codes cannot be completed without a parallel experimental program to supply the new tools with essential data that envelope anticipated design conditions. Data are still needed concerning core physics (critical experiments and differential cross sections, particularly at high burnup), ingress (air/water) phenomena, bypass and lower plenum flow, core and plenum-to-plenum heat transfer, and seismically induced geometry distortion.[b]	Priorities in R&D should emphasize using key test facilities for conducting integral experiments in the HTTF at OSU, refurbish the NSTF at ANL for investigation of water-based ex-core heat removal, perform bypass and air ingress experiments with associated CFD model validation and complete development of 3-D core simulation tools for analyzing complex core behavior under normal and off-normal conditions, including a range of loss-of-forced-cooling events.[a] The development of high-fidelity multi-physics HTGR analysis capabilities on the MOOSE platform is underway.	Low	Necessary R&D to address this topic is well underway. NSTF tests are scheduled for tests through 2021 (depending on validation matrix gaps). Acquisition of other necessary data is underway and approaching completion. Although this topic is an essential licensing concern, the activity is prioritized as low in recognition of the state of R&D accomplished and planned for completion.
3.b	SFR	Complete safety code validation matrices. If existing data is inadequate, identify and design experiments necessary to complete matrices using acceptable scaling practices. The metrics necessary to perform code validations must also be defined.	High	Development and refinement of comprehensive methods that are verified and validated for use in a regulatory safety analysis is necessary to successfully completing a safety review.	High	The SFR R&D analysis codes that exist today have not been reviewed and approved for regulatory use. Depending on design envelope and its relationship to historic information, existing research data may or may not be of sufficient coverage and/or quality to validate their use in all regulatory applications. State of knowledge is considered high but remains contingent on results of a detailed analysis related to legacy data gaps.	Retrieval of operations and safety testing data from EBR-II, FFTF, and TREAT is underway under the DOE-NE ART program. A validation matrix for the SAS4A/SASSYS-1 remains to be developed.[n]	High	Major activity is underway in data retrieval and modernization and verification of SAS4A/SASSYS-1. A code validation text matrix for SFR technology remains to be developed. Given the need for a validated code in SFR design development and licensing, the topic is a high licensing concern.

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
4	**Verify adequacy of evaluation models using an approach conformant with NRC Regulatory Guide 1.203**								
4.a	HTGR	Perform calculations and evaluations of safety-related model adequacy using NRC-accepted validation practices and procedures.[b]	High	The design and safety analysis tool qualification are expected to be done according to accepted regulatory standards. Regulatory Guide 1.203 provides details for adequate assessment when determining the ability of an evaluation model (or its components) to predict behavior (as would be indicated through experimentation).	Medium	AGR safety testing and PIE data acquisition are underway to support fuel performance code validation. Validation experiments are underway for key HTGR fabrication materials (e.g., Alloy 617).[e]	R&D plans include participation in necessary international code benchmark studies. Specifically, the Organization for Economic Co-operation and Development (OECD) MHTGR350 Benchmark of steady state, transient, and lattice codes for prismatic reactors and the International Atomic Energy Agency (IAEA) Uncertainty Analysis Methodologies for High Temperature Reactors.[b] Planned ART R&D is expected to be completed within the next 1-2 years.[k]	Medium	This activity is of medium licensing concern as it is evolving rapidly with respect to expansion in the knowledge base. Regulatory compliant approaches exist to verify the adequacy of HTGR safety models and plans underway to develop products according to those approaches.
4.b	SFR	Perform calculations and adequacy evaluations of SFR safety analysis models using acceptable validation practices and procedures.	High	The qualification of design and safety analysis tools according to regulatory acceptance standards is essential to completing a licensing safety analysis. RG 1.203 specifies that an adequacy assessment be conducted to determine the ability of the evaluation model or its component devices to predict outcomes according to appropriate experimental behavior.	Medium	SFR safety analysis tools that are candidates for use in licensing are either R&D codes or were developed for use by regulatory agencies outside the U.S. These codes may offer promising capabilities, but must be reviewed against applicable NRC guidance and endorsed for domestic licensing use. While there have been prior validation efforts and extensive user histories associated with some codes, important regulatory questions center on what will be required by the NRC to assure acceptance of these tools. The issue will require interactions with NRC staff to address that question.	Various SFR reactor designs have used computer codes maintained by DOE national laboratories. Argonne Computation Code (ARC) systems have been widely used for fast reactor design analysis and consist of a neutronics code suite (MC2-3/DIF3D/REBUS-3/PERSENT), fuel performance analysis code (LIFE-METAL), core deformation analysis code (NUBOW-3D/ANSYS), steady-state thermal-hydraulic analysis code (SE2-ANL), and reactor transient analysis code (SAS4A/SASYS-1).[v] SOFIRE for sodium fire analysis and SWAMM for steam-generator tube rupture assessments are available. Activities to improve and validate legacy tools address NUBOW-3D/ANSYS code benchmark study under the bilateral Civil Nuclear Energy Research and Development Working Group with Japan, updating the metal fuel models of the SAS4A code as part of the PGSFR project with South Korea, validation of the SAS4A/SASYS-1 code through IAEA/CRP FFTF transient benchmark, and validation of the LIFE-METAL code using irradiation data from EBR-II and FFTF as part of the PGSFR project. In addition, BISON (fuel performance) is under development in the Nuclear Energy Advanced Method and Simulation (NEAMS) program. Code analysis capabilities are generally established but simulation of neutronics, thermal, structural, fuel behavior, and hydraulic effects have yet to undergo a rigorous V&V and QA process.[d,i,l,n]	High	Comprehensive lists and crosswalks of needed analytical tools are being established, along with inventories of existing capabilities and associated gaps. These codes can be compared to RG 1.203 and plans established to address regulatory technical requirement deficiencies. While progress has been made over the past two years, the nature and underlying importance of the topic remains a significant licensing concern.

Table 2. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification

a. INL, "Advanced Reactor Technologies High-Temperature Reactor Methods Technical Program Plan," Document ID: PLN-2498, Rev 4, September 29, 2016.
b. INL, "NGNP Program 2013 Status and Path Forward," INL/EXT-14-31035, Rev 0, March 2014.
c. INL, Personal communication with H. Gougar, February 13, 2015.
d. ANL, Personal communication with T. Sofu, March 20, 2015.
e. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2," SAND2012-4260 & SAND2012-4259, May 2012.
f. ANL, "Quality Assurance Program Plan for SFR Metallic Fuel Data Qualification," ANL/NE-16/17, Rev 0, July 5, 2017.
g. GAIN, "GAIN Technology Workshops – Summary Report," INL/LTD-16-39732, August 2016.
h. ORNL, Personal communication with L. Qualls, July 17, 2017.
i. NRC, "Advanced Reactor Research Plan," ML020730737, March 2002.
j. INL, "High Temperature Reactor Research and Development Roadmap," INL/EXT-17-XXXX, DRAFT, April 2017.
k. INL, Personal communication with H. Gougar, June 27, 2017.
l. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel," ANL-ARC-288, June 4, 2014.
m. ANL, "Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors," ANL-SMR-9, May 31, 2014.
n. ANL, "Status of SFR Codes and Methods QA Implementation," ANL-ART-83, January 31, 2017.
o. ORNL, "Qualification of Simulation Software for Safety Assessment of Sodium-Cooled Fast Reactors: Requirements and Recommendations," ORNL/TM-2016/80, April 2016.
p. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014.
q. NRC, "NRC Non-Light Water Reactor Near-Term Implementation Action Plans," ML17165A069, July 2017.
r. ORNL, Personal communication with Brian Ade, January 24, 2017.
s. INL, Personal communication with J. Coleman, February 23, 2017.
t. INL, "Proposed Activities to Address Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation," INL/EXT-16-40668, December 2016.
u. INL, "Graphite Technology Development Plan," PLN-2497, October 4, 2010.
v. ANL, "Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors," ANL/ART-88, Rev 0, April 20, 2017.

Description
<u>Core Heat Removal</u> Advanced reactor R&D must address concerns related to core heat removal and expand those concerns to other issues that may affect plant safety. The role of SSCs important to safety during normal operations (including AOOs), DBEs, DBAs, and BDBEs, and how those SSCs relate to residual core heat removal, must be precisely understood and merged into a safety basis that is comprehensive and credible. High quality research must support a residual core heat removal analysis and thoroughness of the supporting investigation becomes more important the further an individual design moves away from traditional LWR core heat management solutions. For instance, a liquid metal fast reactor operating close to atmospheric pressure and at temperatures far below the boiling point of the coolant will not lead to the same type of depressurization, coolant boiling, and loss of coolant accident (LOCA) experienced by LWRs in the event of coolant leakage or pipe break. This, in turn, could make a traditional LWR emergency core cooling system (i.e., a coolant injection capability under high and low pressure conditions) unnecessary in liquid metal reactors. However, a core heat removal support system is still needed to assure adequate cooling capabilities are maintained during normal and off-normal conditions. In the absence of a demonstrated operational history, comprehensive, high quality R&D programs are expected to fully demonstrate capabilities, capacities, and reliabilities of core heat removal system(s) sufficient (with margin) assure public safety.

Table 3. ART research regarding core heat removal.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	**Confirm reactor core heat removal capabilities**								
1.a	HTGR	Establish a capability to test, evaluate, and validate important design parameters and performance capabilities of the modular HTGR core safety heat removal system. Demonstrate excess heat is transferred to the ultimate heat sink (using air or water as the primary heat transfer medium) at rates adequate to maintain safety. Assessment of system capabilities should consider atmospheric effects, system degradation factors, and system failure potential while in a passive heat removal mode.[a]	High	Modular HTGR designs presume a safety-related passive heat removal system will be employed to ensure core heat is removed during off-normal LBEs. This system is called the Reactor Cavity Cooling System (RCCS).[b] Demonstrating the effectiveness and reliability of the RCCS to operate when required supports the overall safety basis for a simpler and more passive design but NRC requires full qualification of such systems.[a,c]	Medium	Preliminary RCCS designs using air and water as the cooling medium have been developed by HTGR suppliers. Capabilities to prototypically test such systems are limited. Test data is necessary to firmly establish RCCS passive capabilities and provide data for analytic code V&V. While HTGR projects like Fort St Vrain were licensed with safety core cooling system information that is still available, design advancements require further characterizations, resulting in the activity having a medium level of knowledge. Large-scale demonstration of RCCS capabilities is underway at the NSTF at ANL.[d]	Testing of a scaled air-cooled RCCS (based on a GA design) has been completed at ANL's NSTF; a water-based RCCS test plan is currently underway and scheduled for completion in 2019.[d,e] Test scope includes system efficiency, reliability, degradation, weather effects, etc.	Medium	Air-based RCCS testing at NSTF is complete and data analysis. A water-based RCCS test plan is underway. Given the importance of passive RCCS performance in plant safety, licensing success is largely dependent on completion of the NSTF test plan. Licensing priority is medium because critical air-based RCCS testing is done and water-based RCCS is scheduled for completion on a timeline that supports the initial module licensing schedule.

Table 3. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.b	SFR	Develop capability to test, evaluate, and validate key design parameters and performance capabilities for a passive SFR core heat removal system. The system may utilize a compatible substance other than sodium as a cooling medium to direct heat to the ultimate heat sink. Assessments of system capabilities should include consideration of atmospheric effects, system degradation factors, and potential failures in the system.[f]	High	SFR designs being developed will rely on some type of passive heat removal system to ensure core heat remains at safe levels during LBEs. Systems such as these represent a key contributor to the safety basis. This system will undergo assessment during the independent safety review process.	Medium	Although various decay heat removal systems have historically been used in SFRs, the system most likely to be used to address residual core heat removal in a sodium pool-type design arrangement includes multiple loops where each loop consists of a submerged in-vessel DRACS heat exchanger (e.g., twisted tube heat exchanger).[f,g] Similarly, a reactor vessel auxiliary cooling system (RVACS) may be utilized to remove heat from the reactor containment vessel using natural air convection.[c] Physical testing of such systems is limited and additional data reflective of current design trends are likely required to establish performance capabilities and allow V&V of the analytical codes used to assess system performance.	Very compact intermediate heat exchangers and DRACS heat exchangers are desired to reduce overall size of the primary reactor. R&D will bring natural circulation performance of DRACS decay heat removal systems to sufficient levels of maturity to allow use in a sodium reactor environment.[f] Plans that support this R&D effort remain to be established and could look to possible benefit provided by repurposing the RCCS test platform now operating at the ANL NSTF (see Item 1.a above).	High	EBR-II experience and in- and out-of-pile testing shows decay heat removal systems can readily maintain temperatures within design limits for normal and off-normal conditions. Passive decay heat removal will require natural convection cooling capabilities as provided by a DRACS. Until reliability and performance capabilities of such systems can be demonstrated for all design conditions, testing and validating these capabilities represent a key licensing concern. Data obtained from these tests will also be applicable to validation of codes simulating passive heat removal from the vessel.
1.c	MSR	Develop capability to test, evaluate, and validate key design parameters and performance capabilities for a MSR core heat removal system. This system may utilize compatible substances as a cooling medium to direct residual core heat to the ultimate heat sink. Assessments of system capabilities should include consideration of effects arising from perturbations in the ultimate heat sink, system degradation factors, and potential failures in the system.	High	The MSR designs currently proposed are understood to rely on some type of passive heat removal system to ensure residual core heat remains at safe levels during all LBEs. Systems such as these may operate actively or passively and represents a key contributor to the MSR safety basis.	Low	All MSR concepts require attention in the area of passive decay heat removal.[h] Most applied core decay heat removal knowledge today is derived from MSR-E experience and out-of-pile feasibility tests.	General and necessary generic research can be performed in the area of passive heat removal. However, specific designs will require more focused evaluations and confirmatory testing of core heat removal to characterize efficiencies and compatibilities at a representative scale. Establishment of an engineering-scale testing capability that can demonstrate large-scale MSR decay heat removal is projected as necessary within 4-8 years to support technology maturation.[h]	Medium	While the topic of residual core heat removal is a major licensing concern, the potential diversity in heat removal concepts and overall MSR TRL suggests that a generic technology R&D set should be established to focus R&D planning in support of all MSR concepts and licensing safety reviews. Priority is set at medium to recognize generic approaches to topical opportunities are needed to help "focus" R&D planning.

a. INL, "Modular HTGR Safety Basis and Approach," INL/EXT-13-30872, January 2014.
b. INL, "Baseline Concept Description of a Small Modular High Temperature Reactor," INL/EXT-14-31541, Rev 1, May 2014.
c. INL, NRC, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368, Final Report, January 1994.
d. INL, "High Temperature Reactor Research and Development Roadmap," INL/EXT-17-XXXX, DRAFT, April 2017.
e. ANL, Seminar – "Status of RCCS Alliance and Design Planning for Water-based NSTF, Argonne National Lab," February 24, 2015.
f. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel," ANL-ARC-288, June 2014.
g. ANL, "Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors," ANL/ART-88, Rev 0, April 20, 2017.
h. Qualls, A.L, and Hale, R.L., "MSR Technology Roadmap," DRAFT, ORNL/TM-2017/199, Oak Ridge National Laboratory, May 2017.

Description

Material Analysis

Licensing new reactor technology typically depends on the outcome of extensive material science research. New materials used in new applications not previously reviewed or approved by the NRC may require a dedicated R&D program to establish a sound technical basis that supports regulatory approval. That technical basis would evaluate, verify, and confirm material suitability in new applications and include understanding of all plausible modes of degradation and failure. Time-dependent failure criteria for new applications must be developed to assure adequate operational lifespan and sufficient reliability. Development of industry codes and standards by sponsoring consensus organizations, such as the ASME BPV code for advanced reactors are often relied upon to establish a common reference basis for reviewing applications of materials in a structural design approach. The neutron flux, operating temperature, material compatibilities, and corrosive conditions that may accompany new operational environments can challenge existing knowledge limits and raise questions concerning subsequent effectiveness of metals and non-metals used in safety-significant SSCs. The composition of component materials, fabrication and the context of their application, and the resilience to withstand rigors of use must be understood, as well as intrinsic issues like material creep and irradiation effects. Material science is a topic of major focus during a licensing review process and requires answers to questions that can only be addressed through rigorous R&D. It is important to remember that materials research for new reactor applications should be planned and performed according to QA requirements discussed in Section 2.2 of this report and that such research should, wherever possible for purposes of maximizing efficiency, be planned to provide insights for multiple types of reactor design concepts.

Table 4. ART research regarding material analysis.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	Irradiation and property testing of advanced reactor materials								
1.a	HTGR	Generate test data from irradiated nuclear grade graphite samples that more precisely predict material properties and behaviors in support of reactor safety analysis and analytical code development.[a] This information should also support the development of supply sources and qualification of nuclear grade graphite conformant to test data on the HTGR core.	High	Nuclear-grade graphite is used as a moderator and structural component in HTGRs. Graphite preserves core cooling configurations in prismatic block designs. Understanding and predicting irradiation effects for specific grades of graphite to approach its “turn-around” point (transition from shrink behavior to swell behavior) provides insights for selecting appropriate intervals for replacement of graphite internals. This, in turn, affects safety in these applications during normal and off-normal conditions. Understanding graphite behavior is essential to qualify graphite for use in HTGRs. Since “historic” nuclear-grade graphite sources no longer exist, new sources of supply must be developed and qualified for use. Irradiated test data are also necessary to enhance multiscale graphite modeling capabilities.[c]	Medium	Extensive information already exists for historic nuclear-grade graphite. Irradiation-induced creep is currently the primary R&D concern in determining graphite core service behavior. Basic mechanisms of irradiation damage to graphite are well understood, but the magnitude of changes cannot yet be precisely predicted. Since each grade of graphite has a unique structure and texture, additional information is necessary to qualify new sources.	Historic nuclear-grade graphite information for use in HTGRs is available. Recent R&D includes updating ASME BPV Section III, Division 5 code rules for nuclear-grade graphite as new graphite data becomes available.[b] Supplemental graphite irradiation experiments and characterizations are now underway as part of the Advanced Graphite Creep (AGC) program; in addition, chronic and acute graphite oxidation studies are in progress; these tests are detailed in a graphite technology development plan.[c] Data from AGC-1 is currently being added to supplement ASME Section III, Division 5 for graphite internals (Section HHA).	Medium	Although extensive information is currently available, additional qualification test data of nuclear grade graphite is warranted to understand “turn-around points” for different graphite grades and further material behavior predictive capabilities. R&D to qualify new graphite materials and refine analysis tools is currently underway in conjunction with AGR testing. Completion of these tests is important to graphite qualification and future licensing success; licensing priority is “medium” because necessary activities are underway and exert minimal impact on the licensing critical path.

Table 4. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.b	HTGR	The reactor pressure vessel (RPV) must be designed and fabricated to ensure vessel safety functions are adequately maintained during all design conditions. Generate confirmatory data that support related design safety conclusions concerning the RPV.	High	The HTGR RPV is a safety-grade system that provides structural integrity to the core and preserves core-cooling geometry. Regulatory design criteria require the RPV to be constructed of durable materials and compatible with other reactor materials under expected plant conditions. In HTGRs, a key RPV concern involves RPV construction material response to high temperatures.	High	An ART technology development plan outlines the R&D required to design and license a HTGR RPV (assuming SA-508/SA-533 steel is the material of construction).[e] Sufficient data are available to validate mechanical properties of SA-508/SA-533 in this application, but additional data is desirable relative to long-term aging behavior at expected vessel temperatures, as well as to understand environmental effect differences observed from LWR experiences. Data are also needed on the potential effects of impure helium on long-term corrosion and mechanical properties.[c]	As a result of current understanding about supplier design approaches, ART R&D and ASME BPV Section III code development efforts are focused on SA-508/SA-533 for the vessel system (i.e., the reactor pressure vessel, cross vessel, and primary heat exchanger vessel). Alternative materials, such as modified 9Cr-1Mo and 2.25Cr 1Mo steel, are also subject to consideration, but likely will not be used in the initial technology demonstration plant. While current knowledge of SA-508/SA-533 is good, if the design changes to rely on alternative materials, substantial R&D effort may be required to develop qualified data and code information necessary for that material.	Low	Guidance contained in the ASME BPV Sec III code supports current HTGR designs but remain to be endorsed by NRC; endorsement will be an outcome of the first application submitted for review, thereby making this a low licensing priority. Unless designs change to require RPV construction with a material that can withstand higher temps than is now assumed, additional near-term research on this issue is a minor licensing concern.
1.c	SFR	Develop radiation response data for metallic construction materials to adequately support use in SFR fuel cladding and ducts. Perform research to supplement gaps in the existing knowledge base. Ensure data addresses all plant design conditions.[d]	High	All factors that may influence safety performance, integrity, and MSTs during normal and off-normal conditions are evaluated during the regulatory safety review. Objective test data and an ability to predict long-term material performance is necessary to support that evaluation. Thorough knowledge about material properties in applications like fuel cladding and ducting, along with predicted life-cycle performance and quality at point of fabrication, is necessary.	Medium	There are currently two alloy classes with enough radiation response data to be considered in SFR fuel cladding and ducts. Austenitic steel may not be suited to severe irradiation conditions due to void swelling embrittlement. Ferritic-martensitic alloys have the potential to solve irradiation-enhanced swelling, but it is unproven for use in high-radiation conditions like those present in SFRs. Current SFR fuel cladding and duct material knowledge (and fabrication experience) is part of the legacy SFR information bounded by EBR-II and FFTF. These data boundaries may be insufficient for an efficient power generation plant design.[d] Limited data exists concerning material creep rates in advanced reactor environments. Comprehensive gap analysis concerning material property information and materials proposed for use by SFR designs remain to be completed.	While extensive design attention is currently being directed towards oxide dispersion-strengthened ferritic-martensitic alloys that retain swelling resistance and high temperature creep strength, limited amounts of published data concerning this class of material precludes declarations that it is suited for use in SFRs.[d] Research is needed to confirm quality applications of new fuel cladding materials for prospective SFR vendor designs.[e] Perhaps most notable are needed support testing and analysis capabilities for reactor materials that include creep, fatigue, and creep-fatigue tests, material compatibility tests in a high-temperature sodium environment, and fast neutron irradiation tests of reactor internals. A fast neutron source for generating irradiation data has yet to be identified (although projected dpa levels for core support structures, reactor vessel, heat exchangers, and primary piping are modest and may not be critical to licensing). Research results generated through various NEUP and university-led Integrated Research Projects (IRP) may aid the use of ion irradiations as surrogate for fast neutron irradiations.[f]	Medium	All SFR design vendors will need to qualify cladding and duct materials.[d] Additional R&D will be needed to support the issue but that planning is contingent on vendor design choices and scope and quality of legacy SFR data appropriate to that application. Until these gaps are quantified and R&D plans developed to address gaps, which likely could require a fast neutron irradiation experiment to resolve, this activity is a medium licensing concern with potential to become a major licensing issue once application development begins.

Table 4. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.d	MSR	Develop radiation response data for materials used in MSR construction and operation adequate to support their use in the particular MSR concept being considered. Perform research to supplement gaps in existing knowledge base. Ensure data addresses all plant design normal and off-normal conditions.	High	All factors that may influence material performance, integrity, and safety during normal and off-normal conditions will be assessed during a licensing safety review. Objective test data and a demonstrated ability to predict long-term material performance are necessary to support evaluations. Thorough knowledge about material properties in key applications that affect fission product transport, radionuclide barrier performance, predicted life-cycle performance, and quality of safety-significant SSCs, are necessary.	Low	Great variations in MSR design proposals are a barrier identifying specific material irradiation and property testing needs with respect to licensing. Testing methodologies generally remain to be developed "across the board" to assess structural and coolant materials proposed for high temperature nuclear applications.[g] At a minimum, these tests must consider salt-specific changes and interactions with other components both in and outside the fast or thermal neutron radiation environment. Some materials, such as Alloy N, have a body of existing support data, but corrosion and irradiation allowances for coolants and operational design conditions must be determined. Material selection and their status with respect to qualification means R&D should be planned to address as many MSR concepts as possible. It is possible that the lowest-cost, shortest development times will arise through the use of protective cladding over already approved structural materials, but fabrication methodologies and design rules allowing this must be developed, demonstrated, and approved.[h]	Perhaps the most useful test data available is associated with the engineering-scale MSR-E project. ART is preparing to begin material R&D planning within the next year.[g] Salt and material corrosion data over a variety of irradiation conditions for various salt combinations generally need extensive R&D. Fundamental scoping studies on this topic include: fabrication and characterization of salts; salt and material compatibility studies; influence of salt chemistry and impurities; influence of irradiation on salt constituents and material compatibility over time; transmutation and fission product lifecycles for the salt; and development of extrapolation performance models.[h]	Medium	High-temperature use of salts and effects on structural materials in the primary system are a major licensing issue. While it may be possible to qualify materials for limited life (<5 years) in a demonstration reactor to provide operational experience and confirmatory data, the regulatory review for a prototype-scaled design will be challenging unless considerable R&D is initially done. For example, various salt and material combinations may potentially create isotopes with high nuclear cross-section uncertainties. These uncertainties could have safety and performance implications for test reactors. Given the range of MSR concepts is so wide with respect to establishing R&D focus, the issue is assigned a moderate licensing priori**ty pending establishing R&D planning guidance.**
2	**Advanced reactor materials application**								
2.a	HTGR	Ensure data and information is available to define and predict the performance of materials that support the transport of reactor core heat to the heat sink. Include systems like the steam generator, intermediate heat exchanger (IHX), reactor vessel, and other related SSCs.	High	The licensing safety analyst will thoroughly examine the means by which thermal energy generated by the reactor core is transported to external heat sinks during normal and off-normal operations. Only an external (i.e., ultimate) heat sink is credited for plant safety during this safety review. All factors that may influence core heat transfer capabilities during all design facets are to be characterized and justified with supporting data.	Medium	Research objectives related to high-temperature applications of the HTGR steam generator, IHX, the core barrel, and core internals (such as control rod sleeves), are addressed in a technology development plan.[b] The plan was established under NGNP to ensure material performance data are available to develop models that were previously inadequate for certain high temperature alloys that may be added to HTGR codes and standards.	ART activities on this topic are limited pending design decisions on material usage. Improved understanding is needed concerning certain environmental and thermal aging effects of some high-temperature alloys. Welding and joining procedures and certification of components are still needed to address very thick plates and thin sheets. Inspection parameters must be defined and procedures developed. Heat exchange system details and performance requirements cannot be finalized until reactor suppliers specify a required heat load performance envelope.	Low	There are no materials currently recognized as available for use in environments above 800°C (reactor outlet). The allowable life of high-temperature materials is not known to be sufficient to support desired design life.[i] While applicant design choices are needed to support the resolution of this issue, it is understood that near-term designs will remain at or below 800°C, thus making further research on this topic a low licensing priority.

Table 4. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.b	SFR	Establish the experimental and predictive basis that demonstrates safe use of sodium metal in a fast neutron reactor environment. At a minimum, ensure R&D studies address key data gaps for the following phenomena: sodium spray dynamics, sodium jet dynamics, sodium fluid dynamics, sodium pool fire, aerosol dynamics, sodium cavity liner, and sodium-concrete interactions.[d]	High	Using a reactive metallic metal like sodium as a reactor coolant creates significant industrial hazards concerning chemical incompatibility, reactivity, and fire. Historically, the NRC has consistently sought to minimize such hazards to the maximum practical extent. Sodium also creates challenges with respect to testing and inspection of key core components. Technological responses to these new regulatory issues must be sufficient to enable NRC to determine that plant safety will not be unacceptably compromised when using sodium technology.	Low	An expert elicitation of phenomena relevant to sodium technology safety, the criteria important to safety evaluations and the status of phenomena knowledge have been done.[d] Twenty-six gaps of varying degrees of importance were identified in establishing a safety case. While current material performance knowledge may be adequate for important issues on designs close to historic SFR plant operating envelopes, the quality and comprehensiveness of that data remain to be confirmed against emerging designs. NRC listed a number of structural integrity issues during licensing reviews of CRBR and PRISM.[j] Significant research activities will be necessary to develop data that supports predictive effects modeling and evaluations of plant conditions related to those issues and conditions that exceed legacy data envelope boundaries (including DBAs). Thermal aging, sodium compatibility, and irradiation databases could be expanded and high-temperature flaw evaluation methods developed to support licensing and long-term operations.[f]	Facilities exist (or can be reactivated) to support laboratory scale sodium technology tests (including sodium pool fires).[a] New methods and instrumentation must be developed to perform necessary inspections and core tests in the opaque and corrosive sodium environment. Predictive safety analysis tools must be developed. Current sodium technology knowledge is primarily constrained to legacy SFR experience from prior plant operations and information generated from foreign sources. While further R&D of sodium technology is not believed a long lead time item when compared to irradiation testing, the regulatory implications and demonstrations required for metallic sodium use in an SFR requires extensive pre-licensing interaction with NRC, the results of which should guide research planning. Research approaches should be performed in conjunction with NRC staff input.	High	The presence of metallic sodium creates a major source of industrial hazard vulnerability with a significant implication to nuclear safety. Extensive early pre-licensing interaction with the NRC is essential to ensure requisite R&D is planned to adequately address safety concerns. A detailed review of regulatory gaps regarding sodium technology is also recommended. A technology development plan specific to sodium is advised to coherently guide research. R&D related to the safety and use of reactive metals as a core coolant is a major licensing concern.
2.c	MSR	Establish an experimental and predictive basis that demonstrates acceptable use of materials in SSCs in representative MSR operational environments. Ensure material science R&D studies address information gaps for key phenomena (as appropriate to the design) such as: identification and testing of construction materials and their compatibilities; salt chemistry and corrosion in the context of the specific application; chemical and thermally induced changes over time and with changing operational conditions; impact of off-normal event conditions; effect of impurities, etc.	High	Licensing reviews are interested in the material science related to all categories of overall design and particular focus on applications important to safety. Research tests and technical data must objectively support and justify (with margin) the material used and demonstrate their sufficiency over a spectrum of design licensing basis conditions (including DBAs) for the design life of the SSC. Failure to meet this standard will likely result in highly restrictive licensing conditions or denial of a license.	Low	Basic R&D is required for most key MSR construction materials. Structural materials showing limited success at MSR-E are Hastelloy-N and Inconel 600. Used for vessel and piping, Hastelloy-N exhibited low corrosion rates that suggest a prospect for long-term use in SSCs. However, neutron flux to the vessel wall was a significant limit and since both thermal and fast neutrons lead to nickel and lead embrittlement and loss of ductility, issues pertaining to transient response of the material require further study. Similarly, there are multiple intermediate loop coolant salts being considered that can meet both thermal hydraulic and neutronic needs of a design, but must be proven compatible with the multiple materials used in construction and consider salt chemistry changes over time.[h] The overall state of material science knowledge and design rules for MSRs and the subsequent licensing need is low.[g]	Limited material science R&D applicable to MST technology is underway in ART; most information is associated with legacy MSR-E operations. Major studies are needed on basic salt properties that include generation of nuclear cross-section data. Applied research must address materials behaviors within systems, thus suggesting initial work should focus on concept evaluations that are generic to all MSRs. All MSR concepts require studies on salt selection, production, corrosion control, materials qualifications, monitoring, decay heat removal phenomena, and fission product behavior and transport. It is notable that appropriate methodologies must be established to support material science assessment on topics like effect of chemistry changes on corrosion at high temperatures and high velocities over thermal gradients and under varying irradiation conditions.[h]	Medium	The NRC's review emphasizes all aspects of design that may affect safety across the spectrum of postulated design events. Material applications in the context of specific design approaches weigh heavily in licensing success. Confirmatory evidence that material use predictions are accurate and at appropriate scales are a major concern and should be done through integral effects testing. Specific sets of "generic" materials science R&D topics should be developed to guide ART technology development. The activity is given a medium level of licensing concerns and will likely increase once MSR material science R&D boundaries are established.

Table 4. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
3	**Development of material codes and standards**								
3.a	HTGR	Although a license to build and operate a nuclear reactor is granted by the NRC, the construction of key reactor structural components are normally expected to comply with Section III of the ASME BPV code. Ensure the ASME code (or acceptable equivalent) is developed and updated to adequately represent HTGR technology and construction issues (including the RPV).	Medium	Developing and adhering to recognized national standards and consensus codes is important to facilitate efficient licensing safety reviews. However, the NRC is reluctant to endorse new industry codes and standards for reactors in a “piecemeal” fashion and tends to wait until an application is submitted that cites the new code or standard before evaluating and endorsing it. This creates an added burden on the first-of-a-kind reactor technology applicant in that they must: (1) assure codes and standards are appropriately developed for the design; and (2) justify use of the code to NRC during an application review.	High	HTGR consensus codes were initially developed to represent NGNP and are periodically updated; these are contained in Division 5 within Section III of ASME BPV code.[c,k] However, while the code is maintained and relatively current, it has not been reviewed or endorsed by the NRC and is not active for use by applicants. Prospective applicants must review the current state of the proposed standard and confirm it adequately represents their particular HTGR design; deviations must be presented to the standards committee for consensus action.	When HTGR plant design work resumes, reactor suppliers are urged to evaluate current codes and identify additional code support that may be needed from standards development organizations. Appropriate levels of engagement can then be re-established to further update the code. Additional effort to accelerate and maintain momentum in code development will largely be in response to application review schedules.	Low	ASME BPV codes for HTGR application have been proposed and published, but formal approval of the codes needs a vehicle (i.e., an application) to initiate code refinements and a formal NRC review. Additional code work will probably be needed, but resolution is contingent upon HTGR design specifics. Given the current status of application development and needed applicant involvement, this issue is considered a low licensing priority at this time.
3.b	SFR	Although a license to build and operate a reactor is granted by the NRC, construction of key nuclear plant structural components in the U.S. is expected to comply with Section III of the ASME BPV code. Ensure this code and related consensus codes and standards are developed and updated to representatively address the construction of LMR technology and, more specifically, SFR technology.	Medium	Adhering to approved national standards and consensus codes as part of reactor design and construction is important to facilitate efficient NRC safety reviews. The NRC has been reluctant to endorse new reactor codes and standards in “piecemeal” fashion and generally waits until an application citing the new code or standard is submitted. These results in first-of-a-kind reactor technology applicants bearing a major burden in assuring the codes and standards are adequately developed for NRC endorsement.	Medium	LMRs are addressed in the new Division 5 formed within Section III of the ASME BPV code. Rules for SFR construction are also addressed in that section. This code was developed on the basis of an old SFR design approach that may be at significant variance with the design approaches now emerging from prospective SFR suppliers.[i] The code has not been reviewed or endorsed by NRC for general use. Mechanical properties test facilities and analysis capabilities of ANL, INL, and ORNL are available to support ASME code case development and the material compatibility tests in high-temperature sodium environments can be conducted by exposures in the small sodium loops and the Mechanisms Engineering Test Loop (METL) at ANL.	The high-temperature materials and design methods currently contained in the LMR code were developed for SFR license applications, but have not been significantly updated since the 1990s. Modern design methods still need development, R&D demonstration, and incorporated into the code. New materials with enhanced creep strength and life are needed to facilitate specific designs.[i] It is known if ASME code allowables and design parameters could be extended for modified 9Cr-1Mo to support a 60-year design life based on existing data.[f] For early insertion of Alloy 709 into the structural design, ASME code cases could be developed for 100,000-, 300,000-, and 500,000-hour design lives as code qualification data become available.[f]	Medium	ASME BPV codes for SFRs have been proposed but remain to be modified and confirmed as representative of emerging design ideas. A pilot review on consensus codes and standards development has been completed, but an update to the actual codes for SFR use remains to be started.[l] Additional insight into mature SFR designs are likely needed to focus necessary material selections, applications, and qualification efforts. Given the current status and need for applicant involvement, this issue is considered a medium licensing priority that could benefit from ART involvement.

Table 4. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
3.c	MSR	Although a license to build and operate a reactor is granted by the NRC, the construction of key nuclear plant structural components in the U.S. is expected to comply with Section III of the ASME BPV code. Ensure this code or equivalent consensus code is developed and updated to representatively address the construction of MSR technology and represents the specific MSR design concept under review.	Medium	Adhering to approved national standards and consensus codes as part of reactor design and construction is optional to an application but important to facilitate efficient NRC safety reviews. The NRC has been reluctant to endorse new reactor codes and standards in a "piecemeal" fashion and generally waits until an application citing the new code or standard is submitted. This result in the first-of-a-kind reactor technology applicant bearing a major burden in assuring the codes and standards are adequately developed for NRC endorsement and use.	Low	Of the five alloys approved for high temperature nuclear construction in Section III, Division 5 of the ASME Code, none are expected to be able to demonstrate adequate corrosion resistance for MSRs over a typical reactor lifetime of 40-60 years without the addition of active corrosion protection measures. Alternatives include qualifying additional materials, using clad structures, or limiting lifetimes to those allowed by available corrosion effects studies.[h] Relatedly, ASME does not currently have design rules for bi-metallic structures operating at elevated temperatures where differential thermal-expansion-induced creep and fatigue are issues. Material joining technologies, such as brazing, may be considered for high-temperature, low-pressure systems, but are not currently addressed in the code. Rules covering issues like these would need to be developed.[h]	Materials qualified or being qualified for use in other advanced reactors are expected to have limited application when in direct contact with molten salts due to higher corrosion rates or operating temperature challenges. Graphite may be used in FHRs as a moderator, as a particle fuel matrix, and/or as core structural support but qualification of modern grades of graphite for salt service is required.[h] Material selection and qualification options range from using materials that are already code-qualified with a known design methodology and then determining corrosion and irradiation allowances specific to contacting coolants and conditions to the qualification of entirely new materials for service. A consensus standard developed for MSRs at this time would likely heavily draw from MSR-E design and operations experience.	Low	Developing consensus codes and standards is a challenging long-term effort that requires extensive understanding about material use options available to the design being addressed. Given the wide variety of MSR concepts and state of overall TRL associated with this technology, MSR-E experience is likely the only meaningful resource available upon which an ASME code for MSRs can be developed. Given these uncertainties and the maturity of MSR design proposals, the topic is considered a low licensing priority at this time.

a. INL, "Graphite Technology Development Plan," PLN-2497, Rev 1, October 4, 2010.
b. INL, "NGNP Steam Generator and Intermediate Heat Exchanger Materials R&D Plan," PLN-2804, Rev 1, September 23, 2010.
c. INL, "NGNP Reactor Pressure Vessel Materials R&D Plan," PLN-2803, Rev 1, June 14, 2010.
d. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2," SAND2012-4260 & SAND2012-4259, May 2012.
e. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014.
f. ANL, "Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors," ANL/ART-88, Rev 0, April 20, 2017.
g. ORNL, Personal communication with L. Qualls, July 17, 2017.
h. Qualls, A.L, and Hale, R.L., "MSR Technology Roadmap," DRAFT, ORNL/TM-2017/199, Oak Ridge National Laboratory, May 2017.
i. INL, Personal communication with R. Wright, February 27, 2015.
j. NRC, "Pre-application Safety Evaluation Report for the Power Reactor Innovative Small Modular (PRISM) Liquid-Metal Reactor," NUREG-1368, U.S. Nuclear Regulatory Commission, 1994.
k. INL, "NGNP High Temperature Materials White Paper," INL/EXT-09-17187, Rev 1, August 2012.
l. ORNL, Personal communication with G. Flanagan, June 22, 2017.

Description
Instrumentation and Control The coming generation of nuclear reactors will take greater advantage of integrated digital control rooms and utilize automated system diagnostics and responses. Control room staffing requirements will be reduced through automation. Advanced plant designs are expected to push towards a much higher level of automated procedure response in the event of plant upset. Multiple interconnected reactor module plants may share important SSCs and consequently require more sophisticated monitoring, supervisory, and control functions in both primary and support I&C systems. Increased I&C would also be likely where physical interfaces and response capabilities are established between reactor operations and nearby industrial users of direct plant energy. The I&C systems deployed to meet these new goals must be shown to be reliable and precise. They must also support the diagnosis and respond as necessary to normal and off-normal conditions that may affect safety. ART research into digital I&C, such as sensor and control unit development, modernized techniques in data integration, and developing justifications for use of modern integrated core architectures in small modular designs, will be crucial to new I&C systems performance. The results of this research will be subject to analysis and confirmation by NRC on capability, capacity, and reliability during licensing reviews.

Table 5. ART research regarding instrumentation and control.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	Advanced sensors and controls								
1.a	HTGR	Develop in-core detectors and monitoring systems capable of confirming predicted HTGR core operating temperature, power profiles, and fuel operating performance. Systems should be able to detect core irregularities, such as local core hot spots, fuel misloadings, pebble flow anomalies, block-stack motions, and other related conditions.[a]	Medium	NRC staff review of NGNP pre-licensing material concluded that: *"Absent major advances in the development of in-core detector systems for HTGRs, core monitoring and confirmation may have to place significant reliance on near-core and ex-vessel detectors."*[a] This is a preliminary staff perspective that the initial HTGR applicant must address to the satisfaction of the NRC.	Medium	While no full "in-core" monitoring capability has yet been demonstrated as capable over extended periods under expected HTGR core conditions, engineering-scale prototypes have been tested in a relevant environment for lower temperature designs.[b] Approaching in-core instrumentation in a pebble-bed design is particularly problematic due to shifting core characteristics. Reliance on "near-core" detection for higher outlet temperatures may actually add uncertainties regarding actual in-core fuel conditions and could lead to overly restrictive core operating limits when conservatively satisfying functional fuel performance assumptions.	There is no integrated R&D program underway to address this concern. ART has done some technology-specific I&C development like the Johnson Noise Thermometry Monitoring concept, which might be further refined for limited applications in HTGR in-core monitoring. The DOE Nuclear Energy Enabling Technologies (NEET) program is working on high temperature sensors that may offer additional applications, but it is unclear whether that could address HTGR core conditions.[c] High-sensitivity, high-temperature, "micro-pocket" fission chambers and gamma thermometers have been considered as potential options for local power measurements, but only limited work has been done in this area.[d]	Medium	An inability to accurately measure key in-core parameters creates uncertainty regarding presumed validity of predicted analytical safety results. This will likely lead to overly conservative plant operating limits imposed by licensing conditions to add margin in meeting core performance requirements. While a licensing concern, it is considered a medium priority due to relatively low impact on the current licensing timeline. This topic may become a higher future licensing priority.[e]
1.b	HTGR	Develop capability to reliably measure, monitor, and control operation of HTGR RCCS for passive heat removal. The RCCS system typically exhibits low flow and low pressure conditions during both normal and accident plant conditions.	Medium	The RCCS is a system used to maintain plant safety in modular HTGR designs during accident conditions. The system is separate and distinct from the reactor vessel and operates during all modes of plant operation. A capability to predict and monitor system availability and performance is a key safety review issue. The NRC also identified this issue as a general regulatory concern during its review of the GE-PRISM SFR design.[d]	Medium	Until recently, topical knowledge was largely limited to historic information and the capabilities of earlier HTGR designs. Additional RCCS demonstration data has been gathered at ANL's NSTF concerning the air-cooled RCCS; similar demonstrations are about to start at NSTF concerning the water-cooled RCCS approach.[b]	Detailed insights into measurement and monitoring capabilities for air- and water-based types of cooling systems are being developed as a part of the RCCS testing currently underway at NSTF. Finalization of RCCS sensor and control capabilities will be the responsibility of the supplier.	Medium	The RCCS is a critical safety-related system relied upon for residual core heat removal during all DBAs. A competent ability to predict and monitor system availability and performance is a licensability issue for modular HTGRs. While important, the licensing priority is medium in recognition that essential R&D is planned and actively underway on a timeline conducive to licensing.

Table 5. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1.c	SFR	Develop high temperature sensors to reliably measure key safety-related parameters, such as flow and pressure in a liquid metallic sodium environment.[c,g]	Medium	A demonstrated monitoring and measuring capability in challenging environments like liquid metal pools are a concern to safety reviewers. NRC staff noted during review of the GE PRISM design that.”..*establishing and implementing a plan for qualification of a number of sensors expected to be exposed to harsh environments such as reactor cover gas sensors, instruments exposed to primary sodium, and containment instrumentation. This development effort should include conditions for normal operation and accident situations to confirm operability for accident monitoring.”*[f]	Low	The current state of knowledge is limited and mostly a function of historic SFR plant operations. The radiation and high-temperature environment of a SFR suggests remotely operated robotic sensor vehicles are likely needed to perform in-service inspections of reactor and guard vessels in an NRC-licensed plant. While advances have been made in many areas of level-sensing and contaminant detector technology, existing capabilities are insufficient to address the concerns that will arise from NRC regarding I&C and address the lower testing and maintenance requirement goals being set in small commercial SFR designs.[i]	Sensor design and test data from prior SFR operations needs to be collected and assessed to address I&C gaps for emerging designs.[i] There is no integrated effort within ART to address research needs on this topic.[e,l] DOE’s NEET program work on high temperature sensors is available but a specific focus is needed to address the challenges in SFR applications. High-sensitivity, high-temperature, (“micro-pocket”) fission chambers and gamma thermometers have been considered as potential options for localized power measurements, but limited work has been done in this area.[e]	Medium	The inability to accurately measure key parameters in harsh operational environments is a source of licensing uncertainty that undermines confidence in the predictive capacity of assessment models. Plant conditions must be monitored during normal and off-normal operations and sensor reliability and lifespan are less then desirable at this time.[e] While sensor development is needed, the relatively short timeframe associated with this topic in regards to the licensing timeline makes this a medium licensing priority that will likely increase over time.
1.d	SFR	Establish capability to reliably measure and monitor operation of RVACS- or DRACS-type passive heat removal systems in the SFR design. These systems typically operate at low flow and low pressure conditions during both normal and accident plant conditions.	Medium	NRC has indicated that accurately monitoring the SFR’s passive safety cooling system is a significant concern: “*The unusual demands upon the RVACS flow measuring system, as well as its role as a vital safety system component, require that operability checks encompass all operating and accident regimes. Future designs should ensure that testing and calibration for these systems cover all postulated measurement conditions and parameter ranges.”*[f]	Medium	The current state of applied knowledge is based primarily on EBR-II experience and similar (decades-old) in- and out-of-pile testing and analysis and the I&C systems used at that time.[h] This legacy knowledge does not address supervisory circuits and the performance of self-diagnosis capabilities that accompany modern measurement and monitoring equipment. Existing information will not provide the technical basis, which evaluates and justifies currently available industrial component use.	Technical insights into needed measurement and monitoring capabilities for these types of passive cooling systems are being gathered in conjunction with RCCS testing currently underway in the NSTF at ANL. Actual development, testing, and analysis of the I&C needed to apply these insights to resolve RVACS safety concerns is not being addressed by the RCCS testing; ART activities in I&C development on this topic are currently limited.[e]	Medium	The RVACS is a vital safety-related system relied upon for core heat removal during all DBAs. As noted by the NRC, the ability to predict and monitor this vital systems’ performance during DBEs is a licensing requirement for SFRs.[f] Because the topic is not yet seen to adversely impact a licensing timeline, the activity is assigned a medium priority that will likely increase once application starts to be written.
1.e	MSR	Develop sensors and control systems that reliably measure and govern safety-related parameters in a molten salt reactor environment. Sensors and control systems must be compatible with the design attributes of the particular MSR concept.	High	A demonstrated monitoring, measuring, and control capability for safety-significant systems and parameters important to safety is needed for all reactors. These systems must be reliable and compatible with the specific design and support the overall safety basis of the plant.	Medium	Much of the instrumentation needed to monitor and control MSR parameters important to safety will be adapted from existing sensor and control technology as opposed to new development. Basic functions include neutron flux, temperature, pressure, flow, and level measurements. Specialized sensor systems are needed for liquid fissile material inventory, salt system monitoring, and assessing material corrosion. Important salt characteristics must be monitored over time and under changing conditions.[i,j]	Neutron flux measurement instruments (up to 700°C) must be developed for MSR applications. Drift-free, first-principle thermometry is nearing transfer to industry and may have MSR applications.[j] While some instrumentation experience is available from legacy MSR-E operations, this experience does not reflect new I&C design objectives concerning reliability and lifespan. There are no active ART programs underway to directly address salt system monitoring and control.[e]	Medium	Salt system monitoring instrumentation is a specialized measurement technology that needs a dedicated development program different from classical physical process sensor development.[e] A cross-cutting I&C R&D plan is indicated that specifically considers the unique demands of MSR technology. Because the licensing timeline is relatively long for MSRs, the topic is a medium licensing priority.

Table 5. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2	Advanced surveillance and diagnostics								
2.a	HTGR	Develop capabilities to monitor the integrity of reactor internals in modular HTGRs.	Medium	Assuring reactor internal integrity is critical in maintaining core configuration and the geometries that assure a passive cooling capability in the modular HTGR. Maintaining proper core configuration is essential to safety under all design conditions.	Medium	Internal core integrity requirements and methods of confirmation are being assessed as a part of the AGC graphite qualification program conducted at INL.[f] This work also supports ASME code qualification. Results of this work will identify needs for new or additional reactor internal integrity monitoring capabilities.	There is no research currently underway on this topic within the area of I&C development.[d,e]	Low	Additional reactor internals integrity confirmation techniques may be required to be ASME Code qualification efforts. Licensing priority is low until such time as specific licensing need(s) are identified.
2.b	SFR	Develop surveillance diagnostics systems capable of confirming passive feedbacks that may affect plant safety. This system should couple online sensor measurements with computer models and uncertainty propagation to verify that the passive feedback relied upon to prevent core damage in unprotected accidents behave as expected.[c,g]	Medium	Reliance on passive feedback is a key safety characteristic of the SFR safety basis and is relied upon in accident sequences. A capability is necessary to confirm the function and maintenance of this passive safety feature as plant conditions may change.	Medium	The current state of topical knowledge in this area is limited to R&D conducted during past SFR plant operations. A neutron flux monitoring system is required to aid in reactor start-up and efficient plant control, to monitor reactivity changes, and to detect reactor abnormal condition.[h] State of development by SFR technology vendors is unknown.	There is no integrated effort underway or planned to address this research activity. An underlying capability is being developed in the Small Modular Reactor and Light Water Reactor Sustainability programs, but a SFR focus will be necessary to appropriately account for fast reactor specific phenomena, such as core expansion.	Medium	Development of diagnostic capability to assess this core damage prevention measure must be completed to support the SFR's safety basis and licensing. While the topic is important to safety, the level of R&D needed to support a regulatory safety review remains unclear and is a medium licensing priority.
2.c	SFR	Develop methods and capabilities in detecting sodium leakage.[c]	Medium	Liquid metal sodium coolant properties add a dimension of chemical reactivity and material compatibility concern that must be assessed when sodium leaks occur. Understanding and controlling the potential for adverse consequences from sodium leakage must be considered when evaluating SFR reactor safety.	Medium	There is a substantial level of functional knowledge in this area based on historic sodium handling and management techniques.	There is no ART research currently underway in this specific area.	Low	This activity is considered a low near-term licensing priority due to the current state of available and applicable knowledge in this area. Licensing concerns may increase once NRC requirements and expectations on the topic are clarified.
2.d	SFR	Develop reactor internals integrity monitoring capabilities for SFRs.[c]	Medium	Reactor internal integrity must be routinely evaluated and confirmed to satisfy regulatory requirements.	Low	Significant technical challenges exist to routinely perform such monitoring in a sodium pool environment. The current state of knowledge is quite limited; most existing methods are based on LWR environments and likely incompatible for use in liquid metal fast reactors.[h]	I&C technical development activities addressing issues like under-sodium viewing systems that operate in opaque environments is limited within ART. Development of monitoring R&D plans thus far within ART is insufficient to support SFR licensing.[e,l]	Medium	Sodium pool environments are significantly different from the current experience base that must be addressed during licensing. Effective methods for internals integrity monitoring and management remain to be established. Because the issue is not yet seen as impacting the SFR licensing critical path, this is a medium licensing concern likely to increase in the future.

Table 5. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.e	HTGR MSR	Establish reliable and accurate capability for measuring circulating radionuclide activity in the primary cooling loop. Create a capability for monitoring the presence of moisture within the HTGR primary helium loop.	Medium	Radionuclide activity in the primary cooling circuit is a key element of a modular HTGR MST. It will also be an important MST component in MSR technology. Monitoring and measuring this parameter is the purpose of the Specified Acceptable Core Radiological Release Limit (SARRDL), which are being reviewed by the NRC as a licensing component for TRISO-coated fuel via HTGR design criteria guidance.[m] SARRDLs are to be precisely defined, measured, and maintained during normal plant operational conditions.	Medium	Historical measurement of circulating radionuclide activity and moisture was done at the Fort St. Vain plant (e.g., HTGR). It is not known whether heritage methods and measurement capabilities will adequately assess the real-time circulating radionuclide activity measurements expected of SARRDL requirements in modern licensing conditions. MSR technical knowledge on the subject is limited to legacy MSR-E experience with no regulatory insight available on the matter.[e]	There is no ART-related research underway or planned concerning development of SARRDL monitoring capabilities.	Medium	Circulating activity in the primary helium loop is a major contributor to modular HTGR MSTs; it will be similarly important for MSRs. It is unknown how HTGR SARRDL parameters will specifically be addressed in NRC licensing actions, thereby causing regulatory uncertainty as illustrated on Feb 22, 2017, by the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Advanced Reactor Design during advanced reactor design criteria hearings. While an important licensing concern, the issue will require regulatory clarification to ascertain what R&D is still needed for HTGRs. R&D will be needed for MSRs. A medium priority is assigned to this topic.
3	**Human-machine Interface**								
3.a	HTGR SFR MSR	The topics of the human-machine interface (HMI) and control room staffing for modular reactors are generic industry issues consistently identified as regulatory challenges for both integral pressurized water reactors (iPWRs) and advanced reactor technologies; this includes modular HTGRs and SFRs. A plan to address these generic issues is not reviewed in the RTDP pending progress resulting from near-term iPWR licensing interactions. The RTDP will be updated to include these topics in the future once results become publicly known.							NOTE: This item is included as a “placeholder” for future consideration and should be reviewed for regulatory effects as the issue is preliminarily addressed by the iPWR community.

Table 5. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification

a. NRC, "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms (Revision 1)," ML14074A845, Encl. 2, July 17, 2014.

b. INL, "High Temperature Reactor Research and Development Roadmap," INL/EXT-17-XXXX, DRAFT, April 2017.

c. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2," SAND2012-4260 & SAND2012-4259, May 2012.

d. INL, Personal communication with H. Gougar, June 27, 2017.

e. ORNL, Personal communication with D. Holcomb, July 14, 2017.

f. NRC, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368, January 1994.

g. ANL, "Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors," ANL-SMR-9, May 31, 2014.

h. ANL, "Research and Development Roadmaps for Liquid Metal Cooled Fast Reactors," ANL/ART-88, April 20, 2017.

i. INL, "GAIN Technology Workshops, Summary Report," INL/LTD-16-39732, August 2016.

j. Qualls, A.L, and Hale, R.L., "MSR Technology Roadmap," DRAFT, ORNL/TM-2017/199, Oak Ridge National Laboratory, May 2017.

k. INL, "Graphite Technology Development Plan," PLN-2497, Rev. 1, October 4, 2010.

l. ORNL, Personal communication with R. Woods, March 13, 2015.

m. INL, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors," INL/EXT-14-31179, Rev 1, December 2014.

Description
Safeguards and Security: A modern strategy for site security includes "security by design." These considerations must be developed early on in the overall plant design process to be the most effective. Once completed, a more detailed program must be established to guide subsequent design decisions and determine what specific security and safeguards issues need additional developmental attention. Safeguards and security issues that might require R&D include (but are not limited to) new sensor systems, novel approaches in conducting fissile material inventories, and innovative methods in response to security events. A preliminary design security assessment is critical to ensure that effective security and safety measures are integrated into the overall design approach. This assessment also requires appropriate demonstration that a proposed approach can be challenged, tested, and confirmed as appropriate and adequate.

Table 6. ART research regarding safeguards and security.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	**Ensure measures are incorporated that safeguard special nuclear materials**								
1.a	MSR	Develop assessment and protection methods applicable to address the unique circumstances and needs of MSR SNM. Ensure instrumentation and surveillance methods are available to support approaches developed to address MSR SNM safeguards.	High	10 CFR 73.1(a) requires establishment and maintenance of a physical protection system with capabilities for protecting and inventorying SNM at fixed sites and in transit where SNM is used.	Low	Current LWR technology addresses material control and accountability by verifying the physical presence of discrete SNM items, such as fuel rods, fuel pellets, and other nuclear materials present at the site. A salt-cooled MSR may be able to employ many similar techniques, but a salt-fueled MSR will not because the nuclear material is dissolved into the salt. Furthermore, processes like salt clean-up can introduce new opportunities for SNM diversion. Addressing these concerns requires the development of physics-based process monitoring and signature measures, rather than the physical inventory-based process. New approaches in the frequency and intensity of declared and undeclared target SNM inventory control may also be required.[a]	No dedicated ART R&D is underway on this topic. Assessments are initially needed to define a framework that address unique MSR safeguard issues. The assessment would also provide a basis for establishing MSR safeguards objectives, methodologies, and gap identification in measurement and monitoring capabilities. Topics of known current R&D needs include changes in core mass/volume and fuel composition over time, inventory changes during both normal and off-normal conditions, design information verification measures, and direct and indirect methods of material containment and surveillance.[b]	Medium	New safeguard approaches and protocols must be developed that are appropriate to MSR design challenges; the salt-fueled concepts are already recognized as problematic. While SNM safeguards are a critical licensing issue, the licensing priority of this topic is medium until an assessment is conducted to define the gap between needed and available measuring and monitoring needs. R&D planning can then be done once it is informed by the assessment.
1.b	HTGR SFR MSR	Ensure advanced reactor designs incorporate key security and safeguard features that work to affectively lessen security vulnerabilities through integrated design and engineering approaches.	High	10 CFR 73.1(a) requires the establishment and maintenance of a physical protection system with capabilities for protecting SNM at fixed sites and in transit. 10 CFR 73.55 prescribes physical protection requirements for nuclear power reactors. Physical protection includes engineered systems integrated with administrative controls to ensure capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat.	Medium	Regulatory safeguards and security expectations and related implementation solutions for LWR technology is well established and available to non-LWR applicants. While many of the measures do apply to certain non-LWR SSCs, departures may require new design and engineering features to achieve regulatory compliance at the greatest possible efficiency. Evaluating this gap will be a function of the individual designs being considered; gaps are likely to be formidable with respect to salt-fueled MSR concepts.	73 FR 60612 (October 14, 2008) notes that advanced reactor designs should "include considerations for safety and security requirements together in the design process such that security issues (e.g., newly identified threats of terrorist attacks) can be effectively resolved through facility design and engineered security features, and formulation of mitigation measures, with reduced reliance on human actions." These concerns are being formalized by NRC through ten "security design considerations" and can be used as a basis to evaluate requirements and potential gaps in non-LWR technologies and further R&D needs on the topic.[c]	Low	Ten physical and cyber security design considerations are currently proposed by NRC for non-LWRs; these items are (to a great extent) restatements of long-standing existing regulatory safeguards and security requirements.[c] While the considerations could be treated as security design criteria for non-LWRs, the new impact of their finalization on non-LWR R&D and licensing timelines is considered low; security and safeguards by design will be largely based on applicant design decisions.

a. Qualls, A.L, and Hale, R.L., "MSR Technology Roadmap," DRAFT, ORNL/TM-2017/199, Oak Ridge National Laboratory, May 2017.
b. Southern Research, Personal communication with Lance Kim, May 3, 2017.
c. NRC, "DRAFT Non-LWR Physical and Cyber Security Design Considerations – March 20017," ML16305A328.

Description
Accident Sequences and Initiators: This scope includes a variety of complex dynamic systems dealing with diverse fields of investigation such as thermal-fluids, heat transfer, structural, and neutronics modeling capabilities. It also includes the validation basis for simulations. Of particular interest is the availability of validated evaluation tools for accident analysis optimized to assess safety. Additionally, the area addresses potential opportunities for developing computer-based modeling and simulation techniques that improve nuclear safety analysis using high-fidelity, integrated multi-process tools. Using ranges of scenarios and phenomenology identified during safety evaluations, the computer codes and models also encompass analytical capabilities and support data required to assess (with margin) the safety implications of key phenomena and DBEs. Accidents and associated phenomena important to establishing a safety case for a specific design may be insufficiently understood during early phases of technology R&D or described in ways that are not easily translated into risk-informed, performance-based metrics. While a basic level of system design understanding and analysis will be required to support research on accident sequence and initiator topics, it is suggested that ART planning strongly consider technology-inclusive development opportunities that generate a technology neutral perspectives wherever possible. This recommendation targets the broadest-possible benefit to the advanced reactor community while minimizing chances of generating technology-centric limitations like those endemic to the existing LWR-centric regulatory framework.

Table 7. ART research regarding accident sequences and initiators.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1	**Use a regulatory-accepted process for selecting licensing basis events**								
1.a	HTGR SFR MSR	Regulatory requirements concerning the design of new reactors refer to safety evaluations of several different kinds of events within the plant licensing basis, including AOOs, DBEs, postulated DBAs, and BDBEs. Design and design evaluation teams are responsible for selecting LBEs and justifying their selection in terms of risk and assuring safety.	High	The NRC will review and accept the design of LBE selections to characterize the design response and their derivations to assure regulatory requirements are met. A TI-RIPB approach to selecting LBEs is needed for non-LWRs to ensure appropriate sets of limiting events for the technology are properly reflected in DBA selection and that a full set of LBEs that define risk-significant events are applied. It is essential to ensure risk insights are appropriately reflected in the new design and in licensing decisions, including the selection of DBAs.	High	LWRs utilize a strongly deterministic approach in selecting and analyzing licensing basis events; this approach is insufficient for the diversity and novelty associated with non-LWR concepts. A new systematic and reproducible process has been proposed to guide LBE selection; this process is currently being justified to the NRC for adoption as regulatory guidance. While the outcome of this interaction remains to be finalized, the process will be complete, TI-RIPB, and consistent with applicable regulatory requirements.	A modernized TI-RIPB approach to identifying LBEs is currently being considered for adoption as regulatory guidance by the NRC. The process would address a spectrum of events that are to be considered when designing and licensing non-LWRs.[a] The effort is focused on providing developers a robust structure for selecting DBAs to be analyzed in Chapter 15 of a license application. As the proposed LBE selection process is revised and eventually adopted as regulatory guidance, issues may be identified that will significantly impact non-LWR studies and investigations on AOOs, DBEs, BDBEs, and DBAs; these gaps can be used to help identify and prioritize ART R&D planning related to accident sequences and initiators.	Medium	The LBE selection process is primarily a regulatory exercise that uses existing PRA and SSC classification and DID methods; direct R&D support is not needed for that effort. However, as the process is refined and adopted as regulatory guidance, the LBE selection process can be used to guide PRA methods development, SSC classification, and use of DID measures. Deployment of such systems may require key into future R&D planning as a potential “downstream effect,” making this a medium licensing priority with respect to future R&D planning.

a. SC, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors – Selection of Licensing Basis Events,” Southern Company, SC-29980-100, DRAFT Rev A, April 2017.

Description
Probabilistic Risk Assessment: As noted in NRC's "Report to Congress: Advanced Reactor Licensing," in August 2012, advanced reactor designs are to be risk-informed. This makes PRAs a significant component in the overall success of advanced reactor licensing. Reliance on PRAs to inform design approaches and safety decisions allow for the objective identification and incorporation of important safety insights into a design and dose in terms of the actual risk associated with those insights. A typical licensing-related PRA focuses on probabilistic risk and the consequences of occurrence as a result of: (a) the design's robustness, level of DID, and tolerance to severe accident initiated by both internal or external events, and (b) the risk significance of potential human error. Applicants must have adequate technical background information to support safety-related risk assessments to the extent necessary to meet regulatory requirements. Many advanced reactor developers currently face significant challenges in performing a complete PRA due to a lack of historic technical information. Therefore, applications that incorporate new underlying safety hypotheses (e.g., treatment of passive systems), use alternative risk metrics (e.g., core damage frequency or large early release may not be the best figure of merit for a non-LWR design), or have inadequate SSC failure histories (e.g., use new materials and/or SSCs in new safety applications) must develop the requisite support information. While ART R&D may assist in developing certain information essential to performance of a PRA, NRC staff remains the ultimate arbiter when determining whether a particular PRA adequately justifies a risk-based analysis result or safety conclusion.

Table 8. ART research regarding probabilistic risk assessment.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
1		**Develop probabilistic risk assessment approach for use in non-LWR design and licensing**							
1.a	HTGR SFR MSR	Modern advanced reactor licensing approaches involve the development of objective risk-informed data, methodologies, and PRA tools used in predicting safety and security performance. This R&D activity also addresses establishment of quantitative means by which a modernized licensing framework can assess impacts and consequences of significant risk elements without incurring undue conservatism. Probabilistic risk components must be characterized under both normal and off-normal design conditions, as well as under certain BDBE scenarios, and relate directly to the technical safety case in ways that support a comprehensive licensing safety review.	Medium	Developing models for assessing safety-significant risks (with margin) provides a robust venue to demonstrate a comprehensive safety case. Design choices can be justified as well as performance under all design basis conditions that include normal and off-normal operations and associated controls. Because deterministic experience will be largely absent for radical new non-LWR designs, a robust PRA process is needed to provide a sound technical basis for establishing and evaluating the plant safety envelope.	Medium	Although PRA processes and supporting data have expanded and extensively matured for large LWRs and elsewhere, a TI-RIPB framework for non-LWRs faces new technical challenges in the treatment of multi-module plants, the relative lack of relevant historic PRA data, treatment of inherent and passive safety features, and in defining technical adequacy for new systems applications.[a] Guidance for developing PRA models can be found in NUREG-1860. A technology-inclusive approach that introduces and iteratively applies PRAs starting with the early stages of design is now being considered by the NRC for adoption as regulatory guidance.[a] Once regulatory TI-RIPB expectations are nominally set and the development and implementation of safety assessment methods for non-LWRs are understood, new analytic methods or adaptation of traditional PRA methods will likely be required.	ART R&D started work on advanced reactor PRA topics in FY-2013 with work focused on model development, identification of phenomena significant to safety, and evaluation of demonstration problems to establish methods for integrating risk, results, and insights. Examinations were also made about moving beyond current limitations such as static, logic-based models to provide more integrated, scenario-based models based upon predictive tools tied to causal factors.[b,c] Initial demonstrations of these top-level framework attributes was completed in FY-2016 and the tools made available to external stakeholders. No additional research was conducted on this activity in FY-2017.[d]	Low	The need for modified PRA methods applicable to early design stages are an expected outcome of non-LWR licensing framework modernization efforts now underway.[a] This will lead to new tools to better apply and characterize safety insights to a design. The topic is currently considered a low licensing priority until NRC guidance is released (expected in 2019). Once guidance becomes available, gaps in existing PRA data, methods and tools can be more precisely diagnosed and R&D planned to address deficiencies. The licensing priority of this R&D activity will likely increase as suppliers use the new PRA process in early design stages.

Table 8. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance Justification	St. of Know.	State of Knowledge Justification	Research Status	Licensing Priority	Licensing Priority Justification
2.b	HTGR SFR MSR	Enhance seismic probabilistic risk assessment (SPRA) methods and procedures to include SI technology at advanced reactor facilities. Develop nonlinear methods of analysis that can predict responses in nuclear structures equipped with SI and the interaction of nonlinear soils and isolators according to a fully coupled NLSSI analysis.	High	For advanced reactors planning to use SI technologies, current SPRA procedures use scaling assumptions that are not applicable to highly nonlinear systems as would be associated with the use of seismic isolators. New procedures must be available to advanced reactor suppliers and regulators if SI equipment is to be used on key SSCs.[e]	Medium	A ground motion spectral shape anchored to peak ground acceleration is invalid for nonlinear SI bearings. While it is expected that techniques used for conventionally-founded nuclear power plants can be extended to newly isolated facilities, investigations and development is needed to bring these techniques to maturity in SI applications.[f]	There is no research currently ongoing within ART or known ongoing elsewhere on this topic. It is expected that three to five years of moderate R&D effort would be needed to resolve this issue.[g]	Medium	Lack of appropriate SPRA analysis methods will affect SI deployment once SPRA inputs are needed to optimize reactor plant design. While the current licensing timeline suggests a medium-level of licensing concern, this priority will likely increase once an application is started for a non-LWR technology that applies SI measures.

a. SC, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors – Probabilistic Risk Assessment Approach,” Southern Company, SC-29980-101, DRAFT Rev A, June 2017.
b. INL, Summary of “Advanced Small Modular Reactor (SMR) Probabilistic Risk Assessment (PRA) Technical Exchange Meeting,” INL/EXT-13-30170, September 2013.
c. INL, “A Framework to Expand and Advance Probabilistic Risk Assessment to Support Small Modular Reactors,” INL/EXT-12-27345, September 2012.
d. INL, Personnel communication with C. Smith, August 1, 2017.
e. INL, Personal communication with J. Coleman, February 23, 2017.
f. INL, “Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation,” INL/EXT-15-36945, March 2016.
g. INL, “Proposed Activities to Address Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation,” INL/EXT-16-40668, December 2016.

Description
Structural Analysis: Structural analysis tools for LWRs are mature, standardized, and thoroughly validated with extensive data. It is unclear to what extent these same tools can be used in non-LWR plant applications without major modification or additional confirmatory testing. Furthermore, variations between individual advanced reactor technologies may mandate additional tailoring to that technology to address unique plant structural elements that affect safety. The R&D of such updated analysis tools can be planned when specific needs are identified around preliminary stages of plant design. It is recognized that new structural design and analysis capabilities are needed regarding the qualification and use of SI devices at domestic non-LWR facilities, however. Certain advanced reactor designs (such as the modular HTGR) will use deep embedments that shroud the reactor core and heat exchange systems. Additional design considerations may utilize seismic impact dampening systems on key SSCs or even the entire reactor system. Existing seismic designs and seismic safety analysis tools generally presume the facility is located at or close to the nominal level of earthen grade. Since this will not necessarily be the case for some advanced reactors, new analysis approaches will be needed to support seismic isolation equipment design, evaluation, and the impact of below-grade installations and isolation systems appropriate to a safe plant response. These capabilities have not yet been reviewed by the NRC for use in nuclear safety decisions and represent a “generic” R&D opportunity potentially applicable to a variety of advanced reactor technologies. Deployment of SI technology can also expand the range of siting options by enabling higher safety thresholds for locations with more seismic challenges.

Table 9 ART research recommendations regarding structural analysis.

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance (Justification)	St. of Know.	State of Knowledge (Justification)	Research Status	Licensing Priority	Licensing Priority (Justification)
1	Seismic safety technology								
1.a	HTGR SFR MSR	Certified SI design approaches should be enhanced to address base isolation, multi-level embedded foundations, and isolation of major SSCs at non-LWR facilities. Compare site-specific seismic loads to site-independent certified seismic loads over representative case sets where SI technology is deployed and/or foundations are more complex than assumed by accepted regulatory analysis. Develop defined and consistent terminology that supports SI design and licensing.	High	There is no defined pathway for advanced reactor designers to develop and use certified SI systems in nuclear facilities. The little work done thus far has been directed towards SI design certification using demonstration approaches currently tailored to seismically acceptable sites for large LWRs; these approaches are known to be insufficient or not applicable to deeply embedded reactors/associated structures. They are also inadequate for nuclear facilities using SI technology. Task complexity is increased due to confusion on how potentially applicable (legacy) regulatory language is interpreted and applied (e.g., knowing the location of the 0.1g minimum spectrum).	High	Conventionally founded LWR certified seismic designs compare site-specific foundation input response spectrum (FIRS) ground motions to a site-independent certified seismic design response spectrum (CSDRS). This approach must be modified for non-LWR facilities with even simple foundation isolation configurations because in a base-isolated structure, the CSDRS and FIRS are not unambiguously co-located at a single control point. A multi-level embedded foundation and major SSCs that rely on SI equipment to assure safety are beyond the bounds of current regulatory consideration. A technical basis to address SI nomenclature inconsistencies does exist, but a systematic usage structure must be proposed and submitted to the NRC prior to license application development.[a]	The technical basis to address this issue is available and can be developed with limited R&D effort. The largest likely technical challenge will be working through detailed sets of case studies to identify and address ambiguities and technical conflicts that are uncovered. There is no R&D now underway within ART or in the advanced reactor community to address this issue.[b] The effort needed to update the technical basis, complete case studies, and sort out SI-specific terminology could be completed in about a year after work is started; the NRC interactions needed to support topic resolution could extend the projected timeframe.[c]	Medium	The R&D needed to support the issue can be accomplished in a relatively short timeframe with regards to application development. However, because proposed methodologies and SI terminologies must be reviewed, justified, and accepted by the NRC prior to initiating key R&D analysis, the timeline for task completion may be extended. Details of agreement with NRC staff may impact certified seismic motion design efforts for safety-significant SSCs. Thus, a developmental approach must be completed during preliminary design of plants that use a deep embedment and/or SI devices to assure safety.[b]

Table 9. (continued).

ID	Tech.	Research Activity	Reg. Sig.	Regulatory Significance (Justification)	St. of Know.	State of Knowledge (Justification)	Research Status	Licensing Priority	Licensing Priority (Justification)
1.b	HTGR SFR MSR	Performance criteria, analysis tools, and the safety case for SI systems applied to large SSCs must be developed in 2D (horizontal) and 3D (vertical) planes. Innovative isolator design, dampening techniques, and loads at seismic attachment points must be considered in the work. Clarify scenarios where 1D vertical propagating shear wave assumptions cannot be applied with an emphasis on deeply embedded foundations.	High	No performance criteria or applicable regulatory guidance exists for SI technology employed to protect safety-significant equipment and systems and must be developed for endorsement by the NRC. Agreements must be reached concerning the nature of acceptable input and degree of analytical complexity that adequately supports an independent safety evaluation. This issue has been identified by NRC staff as a significant regulatory concern for deeply embedded reactor designs.[b]	Medium	Existing LWR guidance related to foundation isolation provides a starting point for embedded foundation structural analysis. Additional development of the technical basis is necessary to establish a plausible safety case. While a research methodology can be developed straightforwardly, it is currently unclear what these studies might reveal.[a]	Potentially relevant study findings are available concerning the isolation of large equipment in non-nuclear facilities but must be translated for nuclear applications. There is currently no known activity underway to expand the knowledge base for the 2D and 3D scenarios.[b] Some limited research concerning 1D tool development is underway at INL. Seismological inputs to numerical modeling must be developed and could take three to five years to complete.[c]	High	Plants that utilize a deeply embedded basemat (e.g., HTGRs) or SI devices that assure safety require regulatory performance criteria and analysis approaches prior to system design. Vertical propagating shear wave study findings may indicate a need to consider additional seismic load cases in the design. Because these issues must be addressed early in license application development, the topic is assigned a high licensing priority for non-LWRs subject to seismic performance considerations covered by this activity.

a. INL, "Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation," INL/EXT-15-36945, March 2016.
b. INL, Personal communication with J. Coleman, February 23, 2017.
c. INL, "Proposed Activities to Address Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation," INL/EXT-16-40668, December 2016.

Description
Human Factors: Advanced reactor technologies present new operational and maintenance challenges that can be substantially different from current practices. Changes include modernized control rooms with remote supervisory control features and digital I&C functions linked to automated event response programs. Modifications may be made to alarms, control interfaces, and displays that involve SSCs important to safety. New design and safety assurance considerations must be directed towards functional requirements and analysis/performance of such systems and in evaluating the function allocations made to human factors through design. Functional requirements analysis is the identification and analysis of functions that must be performed to satisfy plant safety objectives (i.e., to prevent or mitigate the consequences of postulated accidents). Function allocation analysis considers requirements for plant control and assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and passive, self-controlling phenomena), and (3) combinations of personnel and system elements (e.g., shared control, automatic systems with manual backup). Advanced reactor suppliers are expected to make procedures more computer-based and seek control of safety response actions through automation with operators relegated to a monitoring function capable of bypassing automation when indicated by conditions. New staff training and qualification programs will be needed to maintain these systems. Refocus will be needed on decision-making associated with monitoring and bypass of automatic systems rather than addressing direct control through active operator intervention. Regulatory requirements will demand human factor elements of a new nuclear plant design be assessed with respect to associated risks and consequences; these evaluations are to be supported by objective data and information acquired through dedicated R&D efforts.

Table 10. ART research regarding human factors.

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4. LICENSING PRIORITIES AND RECOMMENDATIONS IN ART RESEARCH

Maturation of modular HTGR, SFR, and MSR technology is linked to ART research activities identified in the Section 3 tables. Successful nuclear plant design licensing, however, must focus on R&D that directly or indirectly relates to public safety. The most important licensing questions directed at R&D programs will likely be related to answering the following:

1. How does the fuel perform during normal and off-normal (accident) conditions, what radionuclides are potentially released during those events, and when are they released?
2. How do radionuclides released initially from the fuel make their way through the plant to the offsite environment?
3. How are safety-related SSCs like systems that remove heat from the fuel designed and maintained to assure acceptable performance when needed to keep radionuclides within the release envelope?

A review of Section 3 activities and opportunities, along with their derived licensing priority estimates, resulted in eight recommendations for near-term ART R&D planning consideration. Implementing these recommendations and continuing to maintain cognizance of long-term licensing needs should enhance prospects for commercial deployments.

The following licensing-oriented R&D recommendations are not presented in rank-order of priority.

4.1 Fuel Performance

RECOMMENDATION 1: Continue recovery, archiving, and configuration control of SFR information from EBR-II and FFTF; preliminarily qualify recovered information according to NQA-1 requirements.

Prospective SFR developers have repeatedly stated in DOE technology working groups and elsewhere that an essential source of experimental information comes from testing experience done at EBR-II (located at INL, operated from 1964-1994 using a metallic core fuel) and FFTF (located at Washington's Hanford Site, operated from 1980-1993 using a mixed-oxide core fuel). Both facilities were designed to demonstrate the viability of sodium-metal cooled fast reactors with the EBR-II design appearing most similar to concepts proposed for nearest-term deployment.[19] Neither of these reactors was licensed under NRC authority (as will be required of new commercial power plants), but the NRC did do a full safety review of FFTF while it was in operation.

Extensive amounts of data, test program information, and findings for EBR-II and FFTF have been recovered by ART. Efforts included salvage of information from data acquisition systems and internal hardcopy reports. With respect to EBR-II legacy data, much collected information has been entered into a modern, configuration-controlled electronic database. Over the last year, ANL has also established a process whereby EBR-II data can be reviewed by subject matter experts for qualification with the NQA-1 requirements.[32] It is recommended that recovery and qualification of this information proceed on an accelerated timeline to enable NRC review of key legacy data and enable subsequent planning to fill gaps that may result in critical safety evaluation information. It is also recommended that formal database configuration controls and data qualification processes be applied to recovered FFTF fuel information as soon as practicable.

It is generally correct to assume that test data and operational information generated by historic DOE reactor technology development projects were generated using good scientific principles and research practices in effect at the time. However, should important information be found deficient in some key quality attribute, an effort may be needed to "upgrade" the dataset using techniques such as confirmatory testing. Because no fast reactor irradiation capability now exists within the U.S. and foreign test platforms may adhere to a lesser level of quality control than required by the NRC, the recovery and qualification of heritage EBR-II and FFTF data and performance of supplemental tests (if necessary) is viewed as crucial to SFR licensing.

It should be underscored that the integrity, quality, and control of recovered safety-related information is as important to NRC safety reviewers as data values themselves. In the absence of appropriate confidence in the data set, NRC staff may reject key information as “unreliable” and require supplemental and/or confirmatory test programs to offset uncertainties. It is therefore imperative that NRC staff be included in early review of heritage data to better ensure legacy information is sufficient for initial licensing purposes.

RECOMMENDATION 2: Identify gaps in SFR metallic fuel knowledge and plan tests to close critical gaps and reduce uncertainties.

Current understanding about metallic fuel, sodium-pool-type SFR design envelopes suggest there may be adequate information available from EBR-II and other heritage facilities to support licensing of a plant similar to the basic EBR-II design.[33] This presumption remains to be fully confirmed with reactor suppliers. Despite this assumption, it is possible that upon review the NRC will conclude insufficient knowledge exists on fuel performance margins outside the normal operational envelope established by EBR-II and FFTF. Since an applicant is required to demonstrate fuel performance well outside the boundaries of normal operation, it is important to maintain a contingency for additional fuel tests in support of licensing. Furthermore, additional fuel irradiations and PIE may be desired by developers to reduce uncertainties and pursue new features that optimize efficiency and enhance performance. Additional PIE on metallic fuel elements irradiated at EBR-II and FFTF can be utilized to fill some of the identified gaps or address concerns that the regulator might have on the quality of specific data. For this purpose, evaluations of EBR-II and FFTF fuel elements for additional PIE will be important.

Fast irradiation facilities are absent within the U.S. and limited at foreign locations. This creates an exceptionally long lead time for supplemental fuel irradiation test programs, thereby making identification and planning to address key fuel qualification gaps a significant licensing concern. Interaction with NRC staff will be necessary to clarify deficient elements and draw conclusions whether historic information adequately addresses safety concerns. Because the metal fuel core variant is most likely to undergo initial licensing review, it is recommended that SFR fuels gap assessments start with metallic fuel and stress fuel performance during all postulated design accident conditions.

RECOMMENDATION 3: Complete Advanced Gas Reactor Fuel Test Plan and the Graphite Technology Development Plan.

Of the advanced non-LWR technologies now supported through ART R&D, modular HTGRs appear closest to license application submission. Significant TRISO-coated particle fuel and core graphite research has been completed over the last decade that will prove critical in licensing. These ART research programs are scheduled to continue for several more years.

The ART VHTR Technology Development Office (TDO) operates the AGR fuel test program to expand the TRISO-coated particle fuel information base concerning: (1) fuel fabrication; (2) fuel and material irradiation; (3) fuel PIE and safety testing; (4) fuel performance modeling; and (5) fission product transport and source term.[21] The qualification approach developed for this fuel (based on AGR test protocols and results) was reviewed by NRC staff and found reasonable with respect to stated objectives.[28] With the conclusion of AGR tests 1, 2, and portions of 3/4, adequate information now exists to develop a “generic” limited scope topical particle fuel qualification report that characterizes:

1. TRISO UCO fuel design characteristics and rationales
2. TRISO UCO fuel product specifications
3. Descriptions of the fuel fabrication process
4. Statistical QA methods that assure specifications are met.

This limited scope report could be written at this time and submitted to the NRC for review and approval by means of a Safety Evaluation Report (SER) issuance. Later, when AGR tests 5/6/7 are completed and combined with design-specific fuel form and core performance information, vendors could complete TRISO particle fuel qualification by adding to the limited scope paper data supplemental information that addresses:

1. Irradiation behavior of fuel (in-pile performance and PIE)
2. Fuel safety test results
3. Establish TRISO UCO fuel performance envelope with failure rates for normal and off-normal conditions.

Completing AGR fuel test program plans through 5/6/7 is a major HTGR licensing concern. Results of AGR tests are essential to qualify coated particle fuel performance, enable fuel performance predictions, and validate MST calculations. It is recommended that a limited scope particle fuel qualification report be developed within the next two years to clarify any remaining fuel qualification issues and reduce uncertainties associated with acceptance of the fuel manufacturing process. A limited scope FQ topical submitted to the NRC in the near future would also capitalize on expertise now available in the AGR program to explain details, protocols, and conclusions of the test program.

Another related effort involves the development of nuclear grade graphite as discussed in PLN-2497, "Graphite Technology Development Plan."25 Graphite is the primary component of the matrix for TRISO-coated fuel particles and constitutes the majority of modular HTGR core volume. The presence and use of graphite in the core substantially influences plant safety and MST calculations. While the basic characteristics of nuclear grade graphite are understood, historic sources of nuclear grade graphite no longer exist. New grades must be established and fabricated by new graphite suppliers, characterized, and irradiated to demonstrate acceptable properties upon which thermomechanical design decisions can be based. Data generated under PLN-2497 (which includes the AGC irradiation experiment program) are essential to establishing the modular HTGR safety basis, and therefore, a key component in future licensing success.

The AGC Program is centered on six capsule irradiations at ATR (designated AGC-1 through AGC-6), followed by PIE of graphite specimens. Development of ASME and American Society for Testing and Materials (ASTM) standards is also essential for HTGR graphite applications. If schedule objectives for near-term commercial deployment are to be achieved, graphite qualification for use in HTGR internals should remain a high R&D priority due to the length and complexity of irradiations and PIE.

RECOMMENDATION 4: Establish the role of fuel in MSR plant safety; develop a definition of fuel qualification appropriate for MSRs and supportive of MST development.

Fuel qualification ensure demonstrates that the fuel will perform (with margin) as expected and required during all design conditions. Defining FQ acceptance criteria for MSR technology starts by understanding the role fuel plays in nuclear safety and establishing performance requirements based on those understandings. Detailed regulatory FQ criteria are not provided under current regulations so proposing requirements and demonstrating their effectiveness is the duty of technology developers.

The role fuel plays in MSR safety likely varies substantially as a function of fuel type (e.g., TRISO, metallic uranium, thorium), fuel form (e.g., solid-fueled or salt-fueled), core design (e.g., fast or thermal neutron), and changes occurring from initial conditions over time and through use. The unique behavior MSR fuels may exhibit during normal and off-normal use may lead licensees to move away from philosophies that consider the fuel itself as the initial “barrier” to fission product release control and towards new performance standards based on fission product control parameters that involve buildup, cleanup, precipitation, retention, and attenuation. MSRs that employ a heterogeneous liquid fuel must also address the formidable challenge of delayed neutron moderation that occur both inside and outside the core.

While MSR-E tests produced extensive information applicable to the FHR design approach, these data offer only a starting point in understanding how MSR fuels may impact safety under various LBE conditions. As such, a working definition about what it means to “qualify MSR fuel” should be established at the start of a MSR fuels qualification program. It is important to include the NRC in these discussions as the staff remains the ultimate arbiter in deciding whether or not a developer has adequately addressed fuel qualification in terms of public safety. It is recommended that efforts be initiated early in R&D planning that lead to clarifications about how MSR fuel qualification criteria are to be established; addressing this issue may require development of regulatory topical report(s) and a formal NRC review and decision document such as an SER.

4.2 Radionuclide Transport Methods

RECOMMENDATION 5: Continue development, qualification, and validation of safety assessment codes and methods compatible with modular HTGR and metallic fuel SFR.

A detailed summary of current VHTR safety assessment tools, modeling capabilities, gaps, and the ART-sponsored R&D underway to address those gaps, is provided in the “Advanced Reactor Technologies High-Temperature Reactor Methods Technical Program Plan.”[26] While the assessment methods and codes being optimized for use in gas-cooled reactors are still “research-level” tools yet to be formally endorsed by the NRC for use in licensing, they are maturing in ways designed to meet regulatory acceptance standards and satisfy requirements for use in safety reviews.

Regulatory Guide 1.203 describes the process that NRC considers acceptable when developing and qualifying nuclear plant design basis evaluation methods and codes. The ART Methods program was planned and is being executed conformant with approaches, practices, and methodologies as set forth in RG 1.203. Continued development of modular HTGR-compatible assessment methods and codes conformant to RG 1.203 is a major concern for licensing. It is strongly recommended that HTGR assessment methods continue advancement according to the criteria and schedules discussed in the ART High-Temperature Reactor Methods Technical Program Plan. (It should be noted that this plan was developed in consultation with affected stakeholders and offers a viable path for addressing key needs of both applicants and NRC concerning nuclear safety assessment tool verification, validation, and quality.)

ANL began work in 2016 to identify and develop safety analysis codes and methods that specifically support metallic fuel SFR safety assessments in response to an expert review panel concern dealing with lack of safety-related SFR code maintenance that was again restated as a 2015 RTDP recommendation.[17,18,34] A parallel support effort led by ORNL aided this activity by identifying software quality assurance (SQA) requirements and best practice options available for development and maintenance of compliance assessment codes.[35]

Because of a broad potential for use in a wide range of steady state and transient safety analyses applications, ANL identified the severe accident analysis systems code SAS4A/SASSYS-1 as a key tool for SFR safety assessments.[36] The SAS4A/SASSYS-1 code was originally developed several decades ago to support DOE research, safety analysis, FFTF assessments, and is still being used today in a research capacity to support conceptual design analysis. The ANL code development team is updating this code to meet SQA best practices (e.g., configuration management, regression testing) and working to identify and address important technical gaps.

Support for SQA needs to continue. In addition, modern SQA practices need to be expanded to other fast-spectrum codes that are required for design and safety analyses. These analyses include cross section processing (MC^2-3), flux and power distribution predictions (DIF3D/VARIANT), gamma heating analyses (GAMSOR), fuel cycle analyses (REBUS), perturbation theory analyses to determine reactivity feedback coefficients (PERSENT), flow distribution calculations (SE2-ANL), fuel performance modeling (LIFE-METAL), mechanistic source term predictions, and containment response (MELCOR).

As development, qualification, and validation progress (using EBR-II, FFTF, and TREAT test data), the fast spectrum safety analysis codes will be available for regulatory use. It is recommended that these and other SFR safety assessment codes continue to be developed by ART. Critical gaps that remain to be addressed include generation of requirements and design documentation, formal code verification and model validation (expected to require a long-lead time to address), use of V&V test suites to improve performance, and creation of documentation that support future commercial use.

4.3 Core Heat Removal

RECOMMENDATION 6: Complete experimental tests at the High Temperature Test Facility (HTTF) and the Natural Convection Shutdown Heat Removal Test Facility (NSTF).

ART VHTR TDO currently supports engineering-scale heat transfer test programs for systems important to safety at the HTTF at OSU and the NSTF at ANL. Completion of planned tests at both operational facilities is recommended to support modular HTGR licensing. The recommendation is made for the following reasons:

HTTF

The HTTF facility was constructed at OSU to simulate HTGR core behaviors while undergoing depressurized conduction cool-down with subsequent air ingress. Facility components are currently configured to replicate prismatic HTGR core conditions; plans have been developed to reconfigure the HTTF core to reproduce pebble-bed core conditions. Non-invasive instrumentation has been added to collect high resolution flow and temperature data (compliant with NQA-1 standards) for computational fluid dynamics code validation.

As a ¼-scale integral experiment, the HTTF can support tests of HTGR fluid behavior during and after depressurization. Results from these tests are crucial to the V&V of safety assessment codes that are needed for licensing. The ART HTR Methods Technical Program Plan[26] details the nature of the HTTF testing program and related separate effects experiments being done by NEUP and other national labs. These efforts to address informational gaps in code development and validation should continue in order to generate essential data needed by both vendors and regulators to assess safety.

The HTTF offers modular HTGR developers a unique R&D platform for adding critical understanding regarding residual core heat removal phenomena. Completing planned experiments at HTTF represents a cost-effective means of addressing important modular HTGR core safety questions.

NSTF

The NSTF is a one-of-a-kind half-scaled engineering test platform built at ANL to originally support simulations of the GE-PRISM (SFR) RVACS. It was refurbished by DOE under the NGNP program to generate NQA-1 compliant data under prototypical modular HTGR vessel heat-up accident conditions. Data collection started in 2014 for a series of air-cooled RCCS experiments; these tests have been completed and an analysis of the findings is underway. The NSTF facility is currently undergoing conversion to support a water-cooled RCCS experimental campaign set to begin in 2018 and continue through 2019.

Information collected from the NSTF tests will yield essential insight into the performance of the safety-related passive vessel heat removal system design unique to modular HTGRs. Some of this data may also be applicable to SFR passive heat removal systems. It is recommended these tests continue to their currently planned conclusion in order to provide the necessary technical basis for a safety review.

Both HTTF and NSTF integral experiments are complemented by separate effects tests being performed at several universities and largely sponsored by DOE's NEUP. Like the HTTF and NSTF facilities, the NEUP-sponsored tests are scaled to one of the HTGR reference designs (i.e., MHTGR, HTR Module, AREVA SC-HTGR) but focus on specific thermal fluid phenomena and conditions. Such data can be used for validating computational fluid dynamics codes.

4.4 Additional Recommendations

RECOMMENDATION 7: Create "generic" R&D activity sets for a "standard" MSR design to increase licensing readiness for the entire technology class.

Current and future MSR technology development opportunities were reviewed in this RTDP. A major observation derived from this review involved the vast range of design options and safety approaches still requiring significant R&D to address licensing requirements. Lack of specificity in design and safety approach represents a significant barrier to performing a meaningful regulatory effects analysis for MSR technology.

Subsection 1.2.3 identifies some of the major variants that collectively fall within the MSR technology class. While all variants still require a major commitment in technology R&D that likely includes construction and operation of a prototype demonstration plant, choosing one MSR design sub-type over another for purposes of ART R&D planning may not be the most effective means by which the technology as a whole can be brought to maturity. It is not the purview of ART to select "winning" MSR safety and design approaches for purposes of R&D planning. It is within the purview of ART to push the technology basis as a whole towards higher levels of technical and licensing readiness. It is therefore logical that ART R&D planning carefully assess the needs of all MSR developers and integrate research planning to provide as much "generic" benefit as is practical. This could be done by establishing a "standard" MSR design envelope from which common sets of research topics can be extracted for scientific investigation and engineering development.

A well-developed general design envelope would enable the creation of a common, high-level safety basis. It would also allow for the construction of conceptual MSTs representative of the bounding parameters of anticipated performance. From this a regulatory effects analysis could be performed as a function of the standard design envelope. Individual reactor developers could benefit from this approach by using elements of the standard design envelope that are relevant to their proprietary approach and departing from standard presumptions as necessary to complement their unique licensing strategy.

With respect to a licensing analysis, a high-level standard design envelope might adhere to the following progression:

1. Decide on the core MSR design concepts to be enveloped by prescribed generic sets of ART R&D guidance
2. Evaluate the significance of fuels, materials, salts and other systems with respect to establishing a fundamental plant safety basis; this could be aided by expert elicitation processes that lead to a requirements-oriented plant PIRT
3. Develop a conceptual (qualitative) MST that represents the generic design performance envelope. Data and information from the MSR-E would be instrumental in creating the first iteration of a “standard” MST for MSRs
4. Begin technology development and evaluate key phenomena and systems necessary to quantify and justify the generic safety basis (i.e., studies in basic core neutronics, thermal hydraulics, fission product transport, etc.).

The current state of MSR-related R&D within ART resides at the first step of this progression. While a small number of DOE research projects like tritium management, TRISO particle fuel qualification, and graphite research may provide relevant information to certain MSR concepts, these are typically accessory benefits derived from R&D specifically targeted at other advanced reactor concepts. Major MSR-specific R&D campaigns dealing with issues like development of in-situ corrosion monitoring systems remain to be undertaken to make the overall technology more relevant and viable.

The diversity in proposed fuels, core designs, and neutron spectrum that comprise the MSR class make it difficult to understand a representative safety case and precisely assess the regulatory effects of technology research. It is recommended that bounding presumptions be established for key design attributes on the basis of plant safety to help guide research planning and aid identification of regulatory impacts associated with that R&D.

A broad pathway leading to MSR prototype licensing is presented in a DOE MSR technology development roadmap.[7] The roadmap identifies the major phases in MSR technology development as:

Phase I: Identification and Engagement of Existing Capability (1-2 years). Evaluate concept maturity and R&D need; engage existing salt science capabilities; measure salt thermal and physical properties; evaluate structural materials and salts; assess corrosion control methods for conventional nuclear alloys; modeling and simulation tool development; cross-section evaluation and measurement; evaluation and development of safeguard protocols; and engagement to develop a licensing framework compatible with MSRs.

Phase II: Establish Larger Scale Testing Capability (2-4 years). Establish new salt science capabilities; corrosion research; tritium management research; review of irradiation test history for candidate salt systems; and initiate new tests as needed.

Phase III: Establish Engineering Scale Testing Capability (4-8 years). Develop engineering scale test loops suitable for reactor pumping and heat exchange testing; and demonstrate large scale decay heat removal.

Phase IV: Enable Technology Development (2-12 years). Component manufacturing technology development; remote inspection, maintenance, replacement, and repair capabilities; specialized sensor development for salt system operation; and salt separation technology.

Phase V: Test/Demonstration Reactor (2025-2030). Initially qualify materials for limited reactor life (possibly 5 years); and develop prototypic licensing strategy.

This RTDP recommendation conforms to roadmap item Phase I "Evaluate concept maturity and R&D need…" and further suggests this evaluation be treated as a precursor to most other ART technology development planning.

RECOMMENDATION 8: Establish cross-cutting I&C system requirements for advanced reactors and develop plan to addresses new performance and reliability needs.

The need to enhance I&C performance, reliability, longevity, and integration is increasing for reactor technology developers. In 2016, ART performed a study that identified high priority I&C development opportunities that support of non-LWRs.[37] Recommendations were generated that recognized shared needs of multiple advanced reactor concepts, as well as the gaps created by the advent of supervisory circuits and self-diagnosis capabilities, now widely incorporated into component architecture. Few of these modern I&C systems have been approved for use in a nuclear plan, thereby making R&D a necessary component for their future use.

The study found modular HTGRs, SFRs, and MSRs all required some level of unique surveillance, monitoring, and control capability to address phenomena like temperature, pressure, flow, and neutron flux. Challenges arise because the instrumentation will sometimes be used at higher temperatures, under corrosive conditions, or will address needs not otherwise encountered in an LWR environment. MSRs offer a particularly noteworthy challenge to I&C developers as new combinations of irradiative and corrosive process measurement must be developed to support salt-cooled operations. Liquid-fueled MSRs in particular will require revolutionary new capabilities for liquid fissile material inventory and tracking (a capability unavailable at this time and likely more complicated than methods currently in use). This makes the development of fissile material safeguards monitoring and surveillance measures a licensing issue for liquid-fueled MSR technology.

Virtually all modern instrumentation incorporates architecture different from that employed in early demonstration plants. Since little regulatory guidance is available for use in addressing this situation, evaluations and technical justifications on use of new systems must be generated at a level of detail adequate to support safety reviews.

Advanced reactor I&C system development appears to be an area where multiple non-LWR types could benefit from a well-planned, cross-cutting R&D campaign. While I&C systems development is not yet considered a top R&D priority with respect to modular HTGR, SFR, and MSR licensing timelines, lack of progress in this area virtually assures I&C development will become a concern in licensing success. It is recommended that ART pursue integrated development of key I&C systems related to advanced reactor plant safety.

5. OTHER ADVANCED REACTOR LICENSING CONCERNS

Two non-LWR reactor concepts discussed in Subsection 1.2 (i.e., modular HTGR and SFR) are approaching the initial stage of license application development while another (i.e., MSR) is far more conceptual in nature and in need of extensive technology development. Indeed, MSRs are likely to employ a prototype plant to support the integrated technology evaluations that will accompany initial unit licensing. As all of these concepts continue to advance towards maturity, new licensing issues with potentially significant R&D implications can emerge as a concern. The following subsections identify topics that are expected to become a future licensing concern, but are not yet a critical RTDP priority recommendation.

5.1 Modernize Regulatory Framework

Non-LWR license applicants historically sought to adapt LWR-oriented requirements for use in a proprietary design. Such an approach relies on exceptions and adaptations to LWR deterministic review standards to address specific technological incompatibilities. The goal was to conservatively bound postulated accident events using the static analysis methodologies being used at the time. The Fort St. Vrain plant in Colorado is an example of this legacy licensing approach.

Suppliers of first-of-a-kind reactors will likely pursue a two-step licensing process based on 10 CFR Part 50. This option requires both a construction permit and a separate operating license. Recently, NRC published implementation action plans aligned with DOE and industry strategies concerning establishment of a more flexible risk-informed, performance-based licensing process for advanced reactors. The work centers on development of technology-inclusive technical licensing requirements and safety design criteria, new approaches to data acquisition, and validation of safety assessment codes.

The TI-RIPB regulatory environment will employ existing PRA processes and be supported by techniques that address safety SSC selection and defense-in-depth measures. Shortcomings will likely be initially faced concerning limited availability in advanced reactor PRA experience, system modeling uncertainties, and questions associated with underlying safety hypotheses, risk metrics, availability of failure data, and treatment of new and relatively unproven inherent and passive safety SSCs. Both advanced reactor license applicants and NRC staff will need sufficient information to justify a decision on safety outcomes.

While NRC is the lead in non-LWR regulation and safety reviews, DOE R&D can support this evolution by:

- Supporting NRC in establishing processes that support a more flexible, risk-informed and performance-based review process (for instance, PRA methods development)
- Supporting NRC development of new licensing requirements appropriate to a new technology
- Supporting NRC in acquiring/developing computer codes and validations
- Developing security measures and material control and accountability safeguard approaches appropriate to the technology
- Resolving technology issues identified in a licensing review (for instance, evaluation of FPT source terms through a molten salt).

Through the GAIN initiative in 2016, DOE entered into a memorandum of understanding with the NRC to work together and assist advanced nuclear technology suppliers to understand and navigate the non-LWR licensing process.[38] Relatedly, this led to a number of advanced reactor technology development and licensing workshops, technology working groups, and the formation of an industry-led Licensing Modernization Project (LMP).[10]

The LMP is currently engaging NRC staff in bringing key proposals to maturity and making them available for licensing use as regulatory technical requirements. Proposals include how LBEs are to be selected, the use of PRA in licensing, how risk-significant SSCs are to be identified and classified, and DID measures appropriate to RIPB decision-making. The LMP is working on a schedule to make these processes available for use by late 2019. If this effort is not completed on a timeline conducive to support the first advanced reactor license application, that applicant will likely revert back to adapting existing deterministic requirements and deal with the substantial uncertainties that accompany such a strategy.

5.2 Develop SFR Consensus Codes and Standards

Voluntary consensus codes and standards provide significant time and cost savings for both the regulator and industry. It does this by improving efficiency, transparency, certainty, and safety. It also adds assurance that regulatory technical requirements are established with a high level of quality. Since most existing regulations, guidance, codes, and standards relate to water-cooled plants, new standards are needed to address coolants, materials, temperatures, operations, testing, maintenance, etc., unique to non-LWRs. Given its place relative to license application development timelines, a concern about codes and standards availability is particularly evident with respect to SFRs.

The NRC incorporates consensus standards by reference. NRC's strategy is to review and regulate technologies like SFRs by working with involved stakeholders to examine currently available codes and standards, identify gaps, and then work with standards development organizations (SDOs) along a path that leads to regulatory endorsement.[39] It usually takes years to develop a consensus and promulgate new or revised regulations founded on those codes and standards. Therefore, directing resources towards SFR codes and standards should be done fairly early during pre-licensing.

It is currently unknown exactly how many standards must be created or amended in order to license SFR technology. However, given the time and effort normally needed to revise, develop, approve, and endorse a revised/new standard, a preliminary scoping review was recently performed at ORNL to gauge efforts necessary to achieve this goal.[40] The pilot study reviewed 865 standards from 30 SDOs cited in 225 different Div 1 Regulatory Guides; additional standards not endorsed for use in a RG were not considered. Of these, 24 consensus standards were recognized to be in need of some degree of revision. An additional 12 new standards would need to be newly created to address topics like the qualification of passive components and use of concrete in high-energy radiation fields.

The basic process for revising or developing a standard consists of:

- Submitting a need and justification for a new or revised standard
- The Standards Committee prepares a draft of the new standard or revision of the existing standard
- The draft standard is issued for internal review and comment
- The Standards Committee revises the proposed standard based on internal reviews and comments
- The draft standard is issued for public comment
- The standard is revised based on public comments
- The Standards Committee submits the revised standard to the Standards Board
- The Standards Board approves the standard for use
- The full consensus standard is submitted (certified) to ANSI for further review and acceptance.

While additional variables such as technical complexity of the new standard and make-up of committees involved in development/approval also affect standard development times, typical time requirements to do this work are on the order of:

- Minor changes to a standard to be approved: 6–24 months
- Significant changes to a standard to be approved: 12–36 months
- Development of a new standard to be approved: 24–60 months.

Staggered submittals will likely be needed to prevent overwhelming individual SDOs.

Initiating development of these standards in the near-term is an activity that will benefit the entire SFR community and could overlap into other non-LWR technologies. In terms of SFR licensing, creating new standards should be the main priority with existing standards requiring significant change being a close second. Delays in addressing this issue can be expected to adversely impact the licensing timeline and potentially delay commercial deployment.

5.3 Fission Product Transport in Metallic Sodium

As a consequence of RTDP recommendations made in 2015, work was started at ANL that culminated in a trial MST calculation (with sensitivity analysis). [33] The calculation used available predictive models and data to identify gaps in current SFR assessment capabilities and the associated informational database. A primary conclusion of the study was that a bounding SFR MST calculation is possible (with certain limitations) utilizing currently available analytical tools and predictive computational models. However, gaps in certain phenomena information created significant uncertainties that necessitated use of conservative assumptions that, if carried into a licensing action, may preclude reduction in plant site boundaries and/or emergency planning zones. Identified gaps (in order of importance based on sensitivity calculations) included bubble transport in sodium, in-pin migration and release, aerosol behavior, hold-up/leakage, vaporization, and dispersion.

A unique feature of pool-type SFR designs involves the physical properties and retention capabilities provided by fully enveloping the core in a dense pool of chemically reactive liquid metal. Metallic sodium has a capacity to significantly influence transport, retention, and scrubbing of radionuclides released from fuel. While it is possible that the first SFR plant can be licensed with a MST developed from available legacy information, the full potential in SFR siting and operation options will not be realized until a more thorough understanding of phenomena affecting fission product transport is developed through experimentation. While extensive sodium metal–radionuclide effects research can be conducted in an ex-core environment, attention must also be directed to the basis of in-core radionuclide generation and related transport phenomena under all operational and design accident conditions.

The testing of dynamic effects for in-pin fission product transport and movement into and through metallic sodium is a significant research opportunity that could greatly enhance commercial deployment opportunities. If additional fission product transport testing is initiated within ART, that research should be scoped in conjunction with supporting fuel qualification activities and analytical safety code development work as these activities are significant contributors to a MST assessment.

5.4 Sodium Technology and Sodium Fire Analysis

The operational history of early SFR plants indicate metallic sodium leaks will occur during normal and off-normal operations. Metallic sodium is highly reactive, potentially corrosive, and in the presence of incompatible constituents like water, generates heat and reaction products hazardous to human health and plant safety. It is therefore essential that these factors be evaluated to assess their precise impacts on the safety basis.

Major questions exist about the reliability of essential components that contact sodium and operate in high temperature corrosive environments. These components include electromagnetic pumps and equipment used for inspection, testing, and maintenance. Instrumentation is needed that can operate in opaque environments and withstand the corrosive effects under all design conditions.

While sodium technology and sodium fire analysis are not yet seen as intruding on the critical licensing path for SFR deployment, these elements must be addressed through appropriate R&D prior to a license safety review.

5.5 MSR Mechanistic Source Term

As stated by NRC in SECY-93-092[42], a MST is to be established by applicants that include analysis of fission product releases based on the amount of cladding damage, fuel damage, core damage, and other factors resulting from the specific accident sequences evaluated. It is to be developed using best-estimate phenomenological models concerning transport of fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers (taking into account mitigation features) and finally into the environment. Development and approval of a new MST has potentially far-reaching implications because it establishes the postulated effects of a reactor facility on the public and surrounding environment. The NRC has consistently indicated that a mechanistic-type of source term is appropriate for evaluation of all advanced reactor designs.[41,42]

An approach for modular HTGR MST development was submitted to NRC staff and found to be "reasonable."[28,31] A source term was to be established by defining the quantities of radionuclides released from the reactor building to the environment during a spectrum of LBEs. Source terms are to be event specific and determined using radionuclide generation and transport models that account for fuel and reactor design characteristics, passive features, and radionuclide release barrier functions. Since the NRC requires sufficient test data to provide confidence in a proposed mechanistic approach, a coated particle fuels development and qualification program was initiated to generate data necessary to understand TRISO-coated particle fuel performance and fission product behaviors. A similar logic was recently applied to a MST "trial run" calculation and found generally adequate for SFR licensing purposes.[19,33,43]

There is no quantitative or quantitative MST currently available for MSR technology *per se.* This is in large part due to a low TRL and wide variations in proposed design approaches and the associated safety basis. Nonetheless, MST development and its demonstrated relationship to safety are extremely important licensing concerns and provide the basis for a regulatory effects analysis. Given the technical complexity and long lead times needed to establish a MST technical justification, source terms development should be considered as a crucial initial concern in MSR fuels and cooling salts R&D planning. Development of assessment tools appropriate to the MSR MST is also essential.

If a "generic" MST model were developed at this time for a "typical" or "standard" MSR design, the MST would likely be based on information gathered from MSR-E tests. From this, details of separate MSR systems could be melded into an integrated and dynamic system that supports basic MST model development and evaluation against regulatory requirements. Regulatory effects reviews could be done using modernized technical requirements of the advanced reactor licensing framework to ascertain "gaps" that exist in the model. From this, near-term R&D priorities could be established and planning done concerning technical evaluations of fuels, materials, and salt properties. Expected R&D opportunities important to MST development include:

- Characterization of salt chemistries and phenomena that affect fission product transport
- Development of the design-specific safety basis
- Fission product and transmutation product lifecycle evaluations, release estimations, and management and mitigation research
- Identification and qualification of materials used in radionuclide barrier construction
- Modeling and simulation tool development concerning assessment of behaviors, safety, and economic performance during normal and off-normal LBE conditions
- Development and scale-up of reactor system component technology such as pumps, heat exchangers, and engineering scale tests used in safety-related SSC development.

For the aforementioned reasons, it is suggested that ART consider establishing a "representative" MST early in the MSR technology development planning process.

5.6 MSR Safeguards Techniques

As was noted in Subsection 3.5, Table 6, liquid-fueled MSRs present a unique challenge in SNM safeguards and accountability. Current SNM safeguards technology addresses material control and accountability by focusing on verifying the physical presence of discrete items such as fuel rods, fuel pellets, and other "countable" nuclear materials present at the site. Salt-fueled reactors will not be able to use these methods because the SNM will be dissolved into a salt, and processes like salt clean-up systems will introduce unconventionalities in current inventory and accounting practices. Addressing this concern requires new physics-based processes and instrumentation based on nuclear signatures for measurement and inventory control.

Assessments are needed to define a framework that can address MSR safeguard issues. The assessment could also provide a basis for establishing SNM safeguard objectives, methodologies, and identify gaps in measurement and monitoring capabilities. Topics of known R&D needs include changes in core mass/volume and fuel composition over time, inventory changes during both normal and off-normal conditions, design information verification measures, and direct and indirect methods of material containment and surveillance. While MSR safeguards R&D are not yet seen as a major licensing priority, the nature of this technology shortfall and the extent to which technique development may be needed makes this topic a significant future licensing issue.

5.7 Research/Test/Prototype Fast Reactor Capability

Most activities identified in the Section 3 tables address a specific constituent or "separate effect" of a larger set of more comprehensive performance requirements. These separations may compel non-LWR developers to consider a research, test, or prototype reactor as a strategy to effectively scale, integrate, and demonstrate test information.

Access to a research- or test-scale reactor of limited thermal power output is likely essential for concepts still needing to perform critical tests and evaluations that cannot be performed in smaller, less specialized facilities. For developers of MSRs that have limited applied experience and hope to submit a license application within ten years, licensing success depends on accessing test loops and simulators that may not be available domestically and of limited foreign availability. While test reactors generally exist to address the needs of thermal reactor developers, there is no domestic test or demonstration reactor capability adequate to serve the needs of fast reactor developers. Proof-of-principle determinations and proof-of-safety demonstrations for components and systems operating in the fast neutron spectrum require test regimes and quality assurance that may not be easily satisfied in a foreign test platform.

In July 2016, an ART planning study reviewed options and priorities relative to establishing a DOE-sponsored advanced demonstration and test reactor (ADTR) facility.[44] The TRL of several reactor types that use coolants other than water was assessed against the following objectives:

1. Deploy a high-temperature process heat option for industrial applications and electricity generation to illustrate the potential nuclear energy offers in reducing the domestic carbon footprint

2. Demonstrate actinide management to extend natural resource utilization and reduce the burden of nuclear waste management

3. Deploy an engineering demonstration reactor for a less-mature reactor technology with the goal of increasing the overall system TRL

4. Establish an irradiation test bed that supports development and qualification of fuels, materials, and other important components/items (e.g., control rods, instrumentation) for both thermal and fast neutron-based Generation IV advanced reactor systems.

The readiness assessment found modular HTGR and SFR technologies have TRLs adequate to support near-term commercial demonstration deployment. The FHR and other similar MSR technologies were less mature and needed R&D and engineering demonstration test facilities. Other non-LWR technology options, such as the GFR, exist at even lower levels of technology readiness.

A licensing analysis was conducted in the ADTR Options Study and will not be repeated here. It was observed, however, that future licensing success of certain non-LWR types like salt-fueled MSRs are highly dependent on irradiation and test loop capabilities that do not currently exist in the U.S. MSR designs uniformly exist at a low TRL right now with only two experimental proof-of-concept versions ever built. Each MSR developer will undoubtedly have a different story concerning their prototype reactor needs as well as the time scale for addressing them. It is important that recommended steps be identified and worked out in conjunction with NRC staff and technology developers in order to let them decide where each thinks they are along the path to deployment as well as define how to get to the end.

While a technology down-select and final decision to proceed with ADTR construction is still pending within DOE, it is clear that developing answers and justifications to key licensablity questions for non-LWRs other than modular HTGRs and SFRs is contingent on timely access to test and demonstration platforms as discussed in the ADTR Options Report.

5.8 Seismic Analysis of Deeply Embedded Structures

Current seismic event response and analysis techniques are linked to presumptions that safety-significant facility SSCs are located at or near the soil surface. However, certain advanced reactor designs, such as the modular HTGR, challenge those presumptions in that they call for the reactor core and heat exchange system to be constructed partially or completely below earthen grade.

Seismic design and analysis licensing rules require SSI to be well characterized for safety-significant SSCs and seismically-isolated equipment. This characterization must be based on validated methods and tools that capably recognize the unique features associated with deep embedments and then analytically predict response to seismic events. Qualified analyses techniques will be needed to design and qualify safety equipment that may include engineered SI devices. Although SI equipment and methods are available to commercial constructions, there is no capability to analyze SI use and consequence effects at a nuclear plant employing deep embedments. Tools must be developed that integrate the seismic, structural, and systems analysis necessary to address unique features like embedments and buried structures.

Since 2015, two studies were completed concerning use of SI in advanced reactors. The first focused on examining the current state of SI technology and identification of regulatory issues, gaps, and technical risks/challenges associated with using SI in advanced reactors.[45] The second study presented a “roadmap” on how those gaps and challenges could be addressed and resolved on a timeline that allows SI at a commercial non-LWR plant.[46] Together, these reports outline specific R&D needs concerning certified design enhancements, SI technology development and testing for 3D seismic ground motions, monitoring and construction, regulatory clarifications, and design performance criteria for analysis tools and large SSCs.

While developing SI technology and seismic analysis tools may be necessary to successfully license some non-LWR concepts, the topic of SI is not yet considered a critical element in licensing success. This is mostly due to a lack of specific design details and uncertainties in a licensing application timeline. Most SI concerns cited in the two INL reports can be addressed within five years by an appropriate R&D

program and interactions with NRC staff. Relatedly, some small modular LWR plants (i.e., the iPWR) are also proposing the use of subsurface embedments; development of proprietary seismic analysis approaches and capabilities may already be underway in support of those technologies. Development of SI technology and analysis tools for non-LWR applications is a R&D capability that already exists within ART and could easily become a licensing priority in the near future.

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