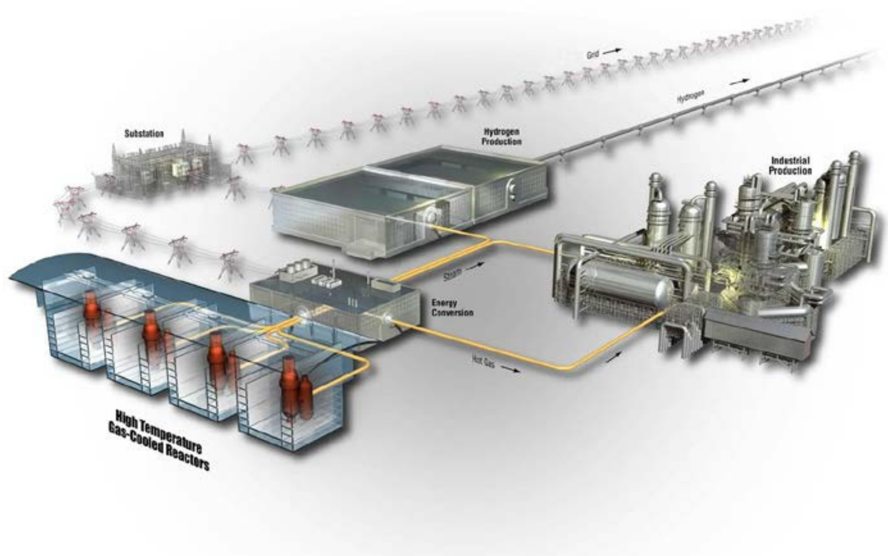
Document ID: PLN-4910
Revision ID: 0
Effective Date: 05/18/2015
INL/EXT-14-32837

Plan

Project No. 29980

Advanced Reactor Technology - Regulatory Technology Development Plan (RTDP)



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ADVANCED REACTOR TECHNOLOGY - GULATORY TECHNOLOGY D VELOPMENT LAN (RTD)	Identifier: PLN-4910 Revision: 0 Effective Date: 05/18/2015 Page i of ix

INL ART TDO Program	Plan	eCR Number 631202

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REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	05/18/2015	All	Newly issued document

SUMMARY

All commercial reactor designs in the U.S. undergo an extensive safety assessment conducted by both the reactor developer and the U.S. Nuclear Regulatory Commission (NRC). As the independent agency responsible for commercial nuclear reactor licensing, the NRC conducts selected confirmatory research, but primarily focuses on evaluating information submitted to the agency in a license application.

The U.S. Department of Energy (DOE) is a major government agency that assists in research and development (R&D) of new reactor technology. A wide variety of tests, studies and investigations may be sponsored by DOE which address key safety system performance parameters and support validation of methods and tools needed by both reactor developers and NRC staff to perform a safety review.

The data and information resulting from DOE-sponsored research is often a key part of the technical development effort needed to successfully license a nuclear plant. Consequently, test plans and conclusions that support a technology safety case and demonstrate regulatory compliance should consider those requirements while protocols are planned and performed. Properly informed planning helps ensure technology research activities adequately address later licensing needs.

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's (DOE-NE) ART program, to key regulatory requirements and licensing challenges likely to affect deployments in the domestic commercial energy market. In response to ART research priorities, the RTDP currently focuses on two technology types likely to undergo NRC safety review within in the next 20 years, i.e., the modular high-temperature gas cooled reactor (HTGR) and the sodium-cooled fast reactor (SFR).

Establishing linkage between reactor research and licensing is complex and requires interaction and coordination with the design community, NRC staff, and researchers working to bring conceptual system designs to maturity. The RTDP was created to aid that linkage and further NRC's Advanced Reactor Policy Statement of 2008 (restated in NRC's 2012 Report to Congress on Advanced Reactor Licensing). This statement encourages reactor research in new safety and security features, or proposals for simplified, inherent, and passive means to accomplish a safety or security function. That information is then to be presented to NRC staff to help assure adequate confirmatory testing, provide for collection of sufficient data to validate computer codes, and show system interaction effects are acceptable.

Section 3 of this document identifies major ART R&D activities concerning modular HTGR and SFR technologies. Insights on the potential regulatory implications associated with these activities are provided in a series of tables. Activities are then analyzed and prioritized with respect to the role they are expected to play in addressing prescribed regulatory criteria and/or developing a safety case. Anticipated lead-times associated with research performance are also considered and activities thought to have very long lead-times or which display major sequential dependencies are noted.

Section 4 contains eight recommendations for ART program consideration. These recommendations, established using information and insights collected from a variety of ART research plans, the ART program leadership in each technical R&D area, and prelicensing precedents with NRC staff, bring attention to topics of current licensing priority. The recommendations consist of:

Recommendation 1: Evaluate, qualify, and control the configuration of historic SFR operations and test data. Two important demonstration plants were decommissioned over two decades ago and recovery of plant information is currently underway. Systematic efforts should be initiated to determine what informational gaps may still exist relative to the current technology safety case, the

quality rigor that can be associated with these historic data, and a configuration management system established to ensure data integrity is not compromised going forward.

Recommendation 2: Planning to address SFR fuel knowledge gaps identified in Recommendation 1 is identified as a licensing priority and should be coordinated with the SFR design community. Fuel tests involving irradiations are often long-lead, yet essential, activities in reactor development. Testing in a fast neutron environment will be challenging because no suitable irradiation capability exists within the U.S. and foreign capabilities are limited.

Recommendation 3: Complete activities described in the Very High Temperature Reactor (VHTR) Advanced Gas Reactor (AGR) Test Plan (PLN-3636) and the Graphite Technology Development Plan (PLN-2497). These plans focus on developing the fuels and nuclear graphite-related information profiles necessary to license a prismatic-block core modular HTGR plant.

Recommendation 4: Complete development of VHTR-compatible safety analysis methods and codes that is already underway. No safety analysis computer codes suited to modern gas-cooled reactor applications have yet been endorsed by NRC for regulatory use. Efforts are now underway within ART to help address this shortcoming and should continue to the planned conclusion.

Recommendation 5: Develop a plan whereby liquid metal fast reactor-compatible safety analysis methods and codes are systematically developed and presented to NRC staff for review. There are no NRC-endorsed safety analysis computer codes currently available that are optimized to the unique elements of SFR technology. Research codes do exist that could be updated and submitted for regulatory acceptance. Efforts required to address this issue are not well understood, however.

Recommendation 6: Test facilities have been established at Oregon State University (i.e., the High Temperature Test Facility, HTTF) and at Argonne National Laboratory (i.e., the Natural Convection Shutdown Heat Removal Test Facility, NSTF) to address HTGR core heat removal. Continuing these (already planned and underway) test programs will produce information essential to support the regulatory safety evaluation process.

Recommendation 7: Form an advanced reactor Industry Advisory Group (IAG) with representatives of the non-light water reactor (non-LWR) design community. Membership would be voluntary and based on interest in ART research. The IAG would be convened by ART project leadership as necessary to provide non-proprietary technical exchange and licensing guidance to ART personnel.

Recommendation 8: Establish a set of fundamental instrumentation and control (I&C) system requirements for advance reactor designs. Creating these requirements will provide guidance to researcher when establishing equipment design/fabrication specifications and testing requirements.

Section 5 identifies additional topics that are expected to emerge as important licensing priorities at a future time. Resolving these issues may require the support of ART research.

It is noted that the RTDP is not a "roadmap" in advanced reactor licensing nor does it replace a design-specific licensing plan. Instead, it seeks to evaluate ART research opportunities and communicate the significance of that research in addressing important safety and regulatory issues. It also assists research planners by drawing greater attention to the needs of applicants and NRC safety reviewers.

The applicant is responsible to write a licensing plan tailored to the design details of the specific technology and commercial offering being developed. The RTDP is a tool that coordinates and guides the regulatory and R&D interface for ART. Thus, the RTDP will be expanded and modified as necessary to meet the needs of ART research and promote the technology development objectives shared by applicants, NRC staff, DOE-NE, and other affected stakeholders.

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TABLES

FIGURES

ACRONYMS

AEC	Atomic Energy Commission
AGC	advanced graphite creep
AGR	Advanced Gas Reactor
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ARDC	advanced reactor design criteria
ART	Advanced Reactor Technologies (program)
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
BDBE	beyond design basis events
BPV	boiler and pressure vessel
CDA	core disruptive accident
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
COL	combined license
CP	construction permit
DBA	design basis accident
DC	design certification
DOE	Department of Energy
DOE-NE	DOE Office of Nuclear Energy, Science, and Technology
dpa	displacement-per-atom rate
DRACS	direct auxiliary cooling system
DTF	designed-to-fail
EBR-II	Experimental Breeder Reactor-II
ESP	early site permit
FCT	Fuel Cycle Technologies (program)
FFTF	Fast Flux Test Facility
FPT	fission product transport
FQ	fuel qualification
FR	Federal Register

HMI	human-machine interface
HTGR	high-temperature gas cooled reactor
HTTF	High Temperature Test Facility
I&C	instrumentation and control
IAG	(advanced reactor) Industry Advisory Group
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
iPWR	integral pressurized water reactor
LBE	licensing basis event
LOCA	loss-of-coolant accident
LMR	liquid metal reactor
LWA	limited work authorization
LWG	(NGNP) Licensing Working Group
LWR	light water reactor
M&TE	measuring and test equipment
MC&A	material control and accountability
MST	mechanistic source terms
MW(t)	megawatt (thermal)
NEET	DOE Nuclear Energy Enabling Technologies (program)
NEUP	Nuclear Energy University Programs
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
NSTF	Natural Convection Shutdown Heat Removal Test Facility
OL	operating license
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PIE	post-irradiation examination
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
QA	quality assurance
QAPD	quality assurance program description
R&D	research and development

RCCS	reactor cavity cooling system
RES	(NRC) Office of Research
RPV	reactor pressure vessel
RG	regulatory guide
RTDP	regulatory technology development plan
RVACS	reactor vessel auxiliary cooling system
SARRDL	specified acceptable radiological release design limit
SFR	sodium-cooled fast reactor
SNL	Sandia National Laboratories
SSC	structures, systems, and component
SSI	soil-structure interaction
TDO	Technology Development Office
TRISO	tri-structural isotropic
TREAT	Transient Reactor Test Facility
TRP	technical review panel
V&V	verification & validation
VHTR	very high temperature reactor

Advanced Reactor Technology - Regulatory Technology Development Plan (RTDP)

1. INTRODUCTION

The U.S. Department of Energy's (DOE) Advanced Reactor Regulatory Technology Development Plan (RTDP) links advanced non-light water nuclear reactor technology development activities sponsored by the DOE Office of Nuclear Energy (DOE-NE) Advanced Reactor Technologies (ART) Program to key regulatory requirements and licensing issues likely to affect new reactor deployments in the domestic commercial energy market. The licensing-oriented discussions and recommendations documented in the plan are not constrained to any particular category, class or type of advanced non-light water reactor (non-LWR) technology, but rather are open to address an array of issues as dictated by contemporary ART research and development (R&D) opportunities. However, because ART research is currently focused upon two specific types of non-LWR reactor concepts, the RTDP is scoped to reflect a similar emphasis.

Within the U.S., nuclear reactors 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 characterizes systems and operations that adequately protect public safety. As a regulatory agency, the NRC does not conduct developmental research on new reactor designs, but rather it focuses on evaluating the information and safety conclusions submitted by an applicant to secure a construction permit (CP), operating license (OL), early site permits (ESP), limited work authorization (LWA), design certification (DC), and/or combined license (COL).

Information required to complete a reactor license application is often generated from sources other than the applicant. As a government agency tasked with performing R&D to assist new reactor technology deployment, DOE-NE sponsors a wide range of studies and technical investigations that provide essential information to the reactor design community. This information may be foundational in understanding system performance, nuclear safety, and component reliability. Accordingly, many of the R&D activities sponsored by DOE-NE should consider NRC policies and regulatory requirements as those activities are initially planned and performed.

The NRC has developed a large body of regulations on the basis of experience gained through large commercial LWR facilities. However, many aspects of those regulations cannot be easily translated to non-LWR applications. To facilitate licensing plants that significantly differ from the large LWR fleet, NRC intends to work on regulatory framework topics as they are identified by prospective applicants and presented to NRC staff. The staff and external stakeholders have already identified significant policy and technical issues associated with small LWR and non-LWR licensing evaluations; these issues, along with their status, can be found in numerous NRC and stakeholder position papers posted on the NRC website. Additionally, the NRC's Office of Research (RES) supports an extensive program that addresses critical areas of anticipatory and confirmatory research in support of the NRC license application review process.

Support of reactor licensing must also consider adoption of applicable quality assurance (QA) requirements that have been endorsed by the NRC. These requirements are applicable to reactor technology research (especially concerning issues important to nuclear safety) as well as the high level design phases of the project. These QA requirements should be implemented within affected research plans using 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" where necessary. Currently, the NRC also allows use of standards described in 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)". Establishing and implementing applicable QA requirements in experimental protocols and system test plans that generate data to be used in safety design decisions is an enabler to success in nuclear plant licensing and commercial deployment.

1.1 Purpose

The RTDP helps identify regulatory issues that may be associated with individual ART R&D activities and directs appropriate planning attention to these issues. The licensing-significant research sponsored by DOE-NE and conducted at national laboratories, universities, and other stakeholder research organizations in support of advanced reactor deployment must meet the same standards of accuracy and quality as other information that is submitted to the NRC by an applicant. Consequently, the regulatory safety criteria that will be leveraged against DOE-sponsored study results must consider and address applicable requirements. However, not all reactor development research carries significant licensing implications.

Figure 1 illustrates how the RTDP can assist in targeting the overlapping interests between ART R&D activities and the NRC regulatory framework.

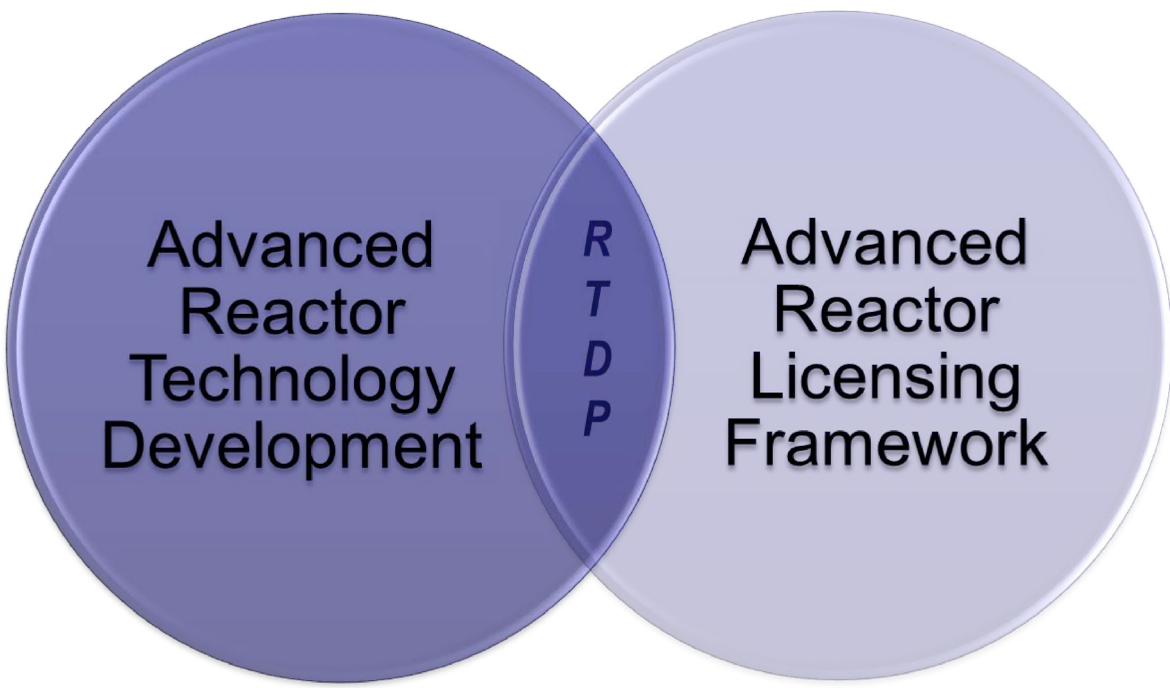

Figure 1. Advanced reactor R&D linkage to licensing framework.

Early and ongoing interactions among reactor designers, NRC staff, and ART researchers may be necessary to clearly establish the influence a particular regulatory criterion may exert on specific research. Consideration of licensing needs during R&D planning has been recognized by NRC as an important aspect in all new reactor technology development programs. 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 testing of new safety or security features that differ from existing designs for operating reactors and/or that use simplified, inherent, and/or passive means to accomplish their safety or security function. Testing should be conducted in ways that 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 encourages design innovations that enhance safety, reliability, and security. However, any technology that utilizes innovations that are not yet proven safe, reliable, and secure must demonstrate that function through a straightforward technology development program. The statement also notes that in the absence of significant operational experience, plans 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 so that NRC staff can assess how the proposed program should be implemented to satisfy associated regulatory requirements.

It is expected that a great deal of design-specific information essential to the conduct of a comprehensive regulatory evaluation will be unavailable during early phases of new reactor technology development. Often, preliminary presumptions about the design safety case must be made to plan tests that support topics like fuel qualification (FQ), mechanistic source terms (MST) development and the qualification of new materials in new systems and applications. Accordingly, the RTDP identifies, assesses, and prioritizes key ART research opportunities with respect to their associated regulatory impact and does so with the goal of assuring DOE technology R&D activities remain coordinated with the technological approaches and licensing strategies used by prospective applicants. It also makes recommendations on research priorities that specifically consider the needs of the NRC independent safety review process.

1.2 Application

Both the NRC and DOE-NE anticipate that significant long-term R&D will be necessary to support non-LWR safety analysis processes. To be more effective in terms of cost and schedule gains, this research should be initially planned and then performed in ways that not only generate information that defends safety decisions for a specific reactor type but also addresses the similar concerns of multiple advanced reactor technologies wherever possible.

An NRC decision to certify a new reactor design and issue a license to build and operate a nuclear plant is guided by scientific and engineering findings that indicate the facility poses acceptable levels of risk to public health and safety, and does not threaten common security. The assessments performed to justify these findings are based on extensive technical evaluations and consequence predictions concerning design safety features, the methods of proposed operation, approaches in accident prevention and consequence mitigation, and barriers that limit radioactive material release to the environment under postulated licensing basis event (LBE) conditions. The offsite radiological doses calculated as a result of bounding release events are a major focus of these safety evaluations.

The criteria currently available for use in evaluating a plant design and the operational data and analysis methods relating to the safety assessment process have been acquired and verified over the last

50 years with a focus on LWRs. The result is a regulatory framework that explicitly incorporates longstanding LWR-centric presumptions concerning reactor safety and analysis technique. Partly as a consequence of these practices, the domestic and international research community has followed a similar pattern in developing and verifying test methodologies, analysis tools, and experimental protocols that are heavily biased towards LWR performance. However, LWR-oriented requirements and analysis tools may not be easily translated to advanced reactor designs or are incompatible with new safety approaches. This difference has been demonstrated in association with the Next Generation Nuclear Plant (NGNP).[2]

Numerous non-LWR concepts are now being proposed with some designs built upon extensive previous efforts in reactor technology development. Some may even benefit from years of prototypical plant operations experience. However, the research base for some “over-the-horizon” reactor concepts now being considered for licensing in 20 or more years is much more limited (or essentially nonexistent). Therefore, it is problematic to try and globally address relational safety approaches and analysis tools necessary to support every non-LWR safety review action that may occur in the future.

In response to this uncertainty, the RTDP is currently focused on two types of advanced reactor technology likely to undergo an NRC licensing action within in the next 20 years; the modular high-temperature gas reactor (HTGR) and sodium-cooled fast reactor (SFR). A variety of ART research activities on these concepts are actively underway at this time. This research is intended to be consistent with the objectives expressed by NRC in the 2012 “Report to Congress: Advanced Reactor Licensing”. Additional types of advanced reactor design may be added to the RTDP in response to expansions in the direction of ART research.

2. SCOPE OF TECHNOLOGY-ENABLING RESEARCH

This RTDP identifies areas of regulatory concern relative to technology-enabling R&D (as sponsored by the DOE-NE under the ART program) that will eventually lead to commercial deployment of advanced reactors within the U.S. An analysis scheme has been devised whereby licensing perspectives can be overlain upon individual ART research activity descriptions so that applicable regulatory framework elements can be assessed and prioritized during R&D planning and performance. The RTDP informs research planners by offering a licensing perspective on each activity and proposing key recommendations based upon the NRC safety review process, the state of topical knowledge, and the expected sequence of research activities that might combine to adversely impact a critical path in licensing success.

As R&D activities important to plant safety are planned and performed, it may become necessary to establish a prelicensing dialog between various research stakeholders. These interactions may focus on soliciting ideas and inputs from technology developers, reactor design vendors, NRC staff, and other entities concerning requirements, determining whether proposed approaches are realistic in addressing a particular issue, and communicating test outcomes and conclusions.

Eventually, a licensing plan will be developed for each reactor design. This plan solidifies the design-specific strategies used in addressing regulatory criteria and couples them to specific plant design features and safety goals. The licensing plan must also ensure that appropriate safety analysis methods and computational tools are available that can demonstrate attainment of goals and criteria. However, until a licensing plan is established to guide this interface, the insights necessary to inform advanced reactor technology researchers (and possibly NRC staff) of pertinent licensing issues can be addressed by the ART RTDP.

2.1 Key Research Areas

Many kinds of R&D will be necessary to establish the safety basis of a new reactor design. Analytical safety tools can become a significant licensing obstacle if not considered and addressed during early R&D planning. If appropriate codes are unavailable or if their validity cannot be confirmed to a degree that supports conservative conclusions about plant safety, a license may not be granted. At a minimum, analytical tools must always be able to verify the adequacy of specific design features that ensure adequate heat removal from the core, maintain reactivity control, and provide for radionuclide retention.

A regulatory safety analysis encompasses the areas of accident analysis and reactor and plant analysis. Reactor and plant analysis measures reactor and plant performance under normal operating conditions, whereas accident analysis verifies reactor and plant performance under design-basis conditions. Both areas of analysis rely on thermal-hydraulic (or in the case of non-water technologies, thermal-fluid) and neutronic (reactor physics) aspects of a technology. Major 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.

As already mentioned, every licensed reactor is required to have appropriate methodologies, analytical tools, and high-quality support data available to use when addressing plausible questions about safety challenge. These challenges are also generally organized according to the three basic functions of:

1. Adequate core heat removal

   Challenges to heat removal involve timely and sufficient cooling of fuel elements, the core, the reactor vessel, and design elements used for radionuclide retention. These elements are presumed critical to preventing fission product barrier failures. Assuring fission product barrier integrity is a critical safety priority. Back-up systems may be necessary to provide adequate defense in depth to ensure that required safety functions are performed during anticipated conditions.

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 to handle a challenge. It must be demonstrated that reactivity control features will perform as intended in all circumstances where the function is essential to maintain safety.

3. Control of radionuclide release

   Challenges to retention of radionuclides involve maintaining fuel integrity, core structures, and other barriers relied upon to limit releases of radioactivity to the environment.

It should be noted that any reactor technology that uses a highly innovative fuel (e.g., reactor fuels containing thorium) and/or new methods to assure reactor core cooling (e.g., molten salt as a heat transfer fluid) in combination with other new active or passive safety features must still address the basic elements of the existing safety analysis process (i.e., thermal-fluids behavior, neutronics, fission product behavior).

A diagram of some major research areas in relation to the plant safety review process and licensing is provided in Figure 2.

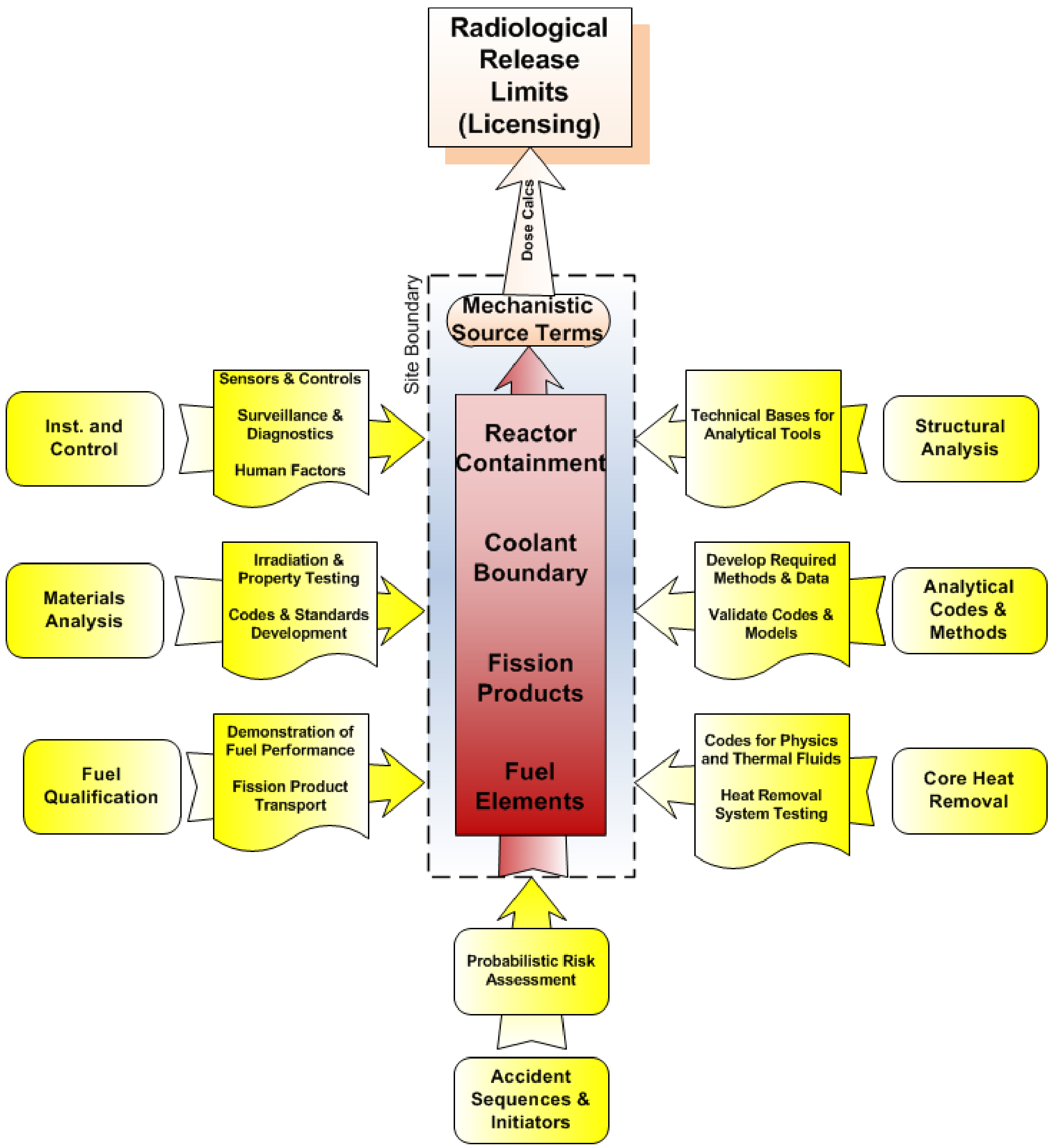

Figure 2. R&D elements that contribute to the plant safety and licensing review.

Realistic yet conservative radionuclide release analyses of all factors affecting dose calculations are essential to a positive safety review outcome. This analysis must be based on objective test information concerning fuel behavior during normal and off-normal conditions. Fission product release and transport characteristics must be understood for bounding design conditions and meet applicable radiological release limits for those conditions.

The following subsections further discuss the role technology R&D contributes in addressing advanced reactor licensing issues.

2.1.1 Fuel Qualification

The design, manufacture, and use of nuclear fuel are foundational to plant safety. Extensive fuel test knowledge and characterization data is required to meet established regulatory criteria. Because the infrastructure required to collect key fuels-related data is often highly specialized and relatively scarce, DOE is a leading resource in fuels-related research. A FQ program that includes long-term irradiation tests will generally be necessary to fully evaluate new and modified fuels. These factors, along with the long lead-times needed to support certain types of in-core fuel tests, typically causes fuels research to be a significant licensing concern.

A robust experimental database is necessary to understand fuel system responses to a range of design and burnup conditions. Simulating fuel performance and fission product transport (FPT), retention, and releases under accident conditions also relate to this topic. The licensing analysis of ART research regarding fuel qualification is provided in Table 1. Licensing recommendations specific to FQ are provided in Subsection 4.1.

2.1.2 Mechanistic Source Terms

A source term refers to the release of radionuclides from the fuel to the plant and beyond to the environment. With respect to advanced reactors, stakeholders (including the NRC) recognize that a "mechanistic source terms" approach should be employed. The MST focuses on realistically modeling the release and transport of radionuclides from the source to the environment for specific scenarios while accounting for retention and/or transmutation phenomena and uncertainties associated with the process. Determining an MST for radionuclide transport that involves complex phenomena requires extensive test-based knowledge and a well-developed modeling capability for all involved processes of significance. While development of a detailed and technically sound MST will be design-specific and is ultimately the responsibility of applicants, the approaches, tools and methods used to perform safety assessments of MST-related process may be useful over a range of differing design concepts.

Radionuclide releases must be defined at the source (i.e., the fuel) and quantified with respect to transport behaviors and attenuation factors as paths are established to the environment. Concentrations of radionuclides retained behind radiological release barriers (as a function of time) are crucial in defining an acceptable MST. Release and transport of key fission products during LBEs may need to be addressed at least in part through fuel testing. The goal underlying all such tests is to obtain a quantitative understanding of the relevant phenomenology and enable consequence predictions concerning released fission products.

R&D efforts related to MST development are closely related to and highly reliant upon activities that qualify reactor fuel. This is largely because reactor fuel types and performance data provide a starting point in the MST analysis. The licensing analysis conducted relative to ART research gaps in MST development is provided in Table 1 (in conjunction with FQ information). Recommendations that affect MST research are discussed in Subsections 4.1.

2.1.3 Analytical Codes and Methods

As already noted, developing, refining, verifying and validating (V&V) analytical methods is essential to the safety analysis process. Analytical techniques must be available to support the model and enable prediction of important phenomena. Often these phenomena are initially identified through expert elicitations. Once the candidate parameter is identified, V&V plans can be developed that produce data concerning the parameter.

The licensing analysis related to ART research in analytical codes and methods is provided in Table 2. Recommendations that focus on this research topic are provided in Subsection 4.2.

2.1.4 Core Heat Removal

Advanced reactor research must address a variety of issues regarding core heat removal. This topic is closely related to core design and can touch on any structure, system, and component (SSC) that is relied upon to provide a heat removal function during anticipated operational occurrences (AOO), design basis accidents (DBA), and beyond design basis events (BDBE). How those elements relate to safety must then be precisely understood and merged into an overall plant safety basis. New research information that supports a heat removal analysis becomes more important as advanced reactor technologies pursue more passive methods of heat removal.

The licensing analysis related to ART research activity regarding core heat removal and associated core design issues is provided in Table 3. Recommendations that affect this topical area are provided in Subsection 4.3.

2.1.5 Materials Analysis

A sound technical basis is required to enable evaluation and confirmation of the integrity and modes of failure in safety SSCs that use new materials or materials in new applications. Time-dependent failure criteria for these materials must be developed to ensure safety is maintained and demonstrate adequate operational life. Development of common standards concerning material adequacy and applicability R&D 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 such organizations are recognized by NRC as a means to confirm structural materials design is technically sound and appropriate.

For non-LWR technologies, particularly technologies geared towards elevated temperature applications or which employ a novel cooling medium such as liquid metals or fluorine eutectic salts, a materials performance database may not exist to support a safety analysis. Time-dependent failure criteria for a material in high-temperature or corrosive environments may need to be established to ensure adequate performance during the operational life of the component.

Confirmatory analytical tools and predictive performance models are also a key part of material science. Likely areas of study include initial material behavior before and after fabrication, effects of irradiation on material properties, aging in a radiation environment, and the corrosion behavior of structural materials under various plant conditions. A scarcity of operating experience for most advanced reactor types, coupled with the potential for high-temperature and/or corrosive operational conditions, makes this topic a consistent concern during licensing.

The licensing analysis related to current or planned ART research regarding materials is provided in Table 4. Recommendations related to ART research that may affect this topic are provided in Section 4.

2.1.6 Instrumentation and Control

Advanced reactor developers need to ensure that the instrumentation and control (I&C) systems deployed in the future can adequately measure, diagnose, and respond to safety and operating parameters relative to conditions deemed important to safety. Such configurations may entail new I&C requirements, new sensor types, updated data integration techniques, and first-of-a-kind displays. Some I&C system sensors will operate in environmental conditions significantly different and possibly far harsher than experienced in the current LWR fleet. Temperature, pressure, flow, and neutron instrumentation may need to operate in higher temperatures or corrosive environments. Combinations of high temperatures and chemically reactive process environments (as might be found in liquid metal fast reactors) can create significant challenges to instrument design. Severe environments may also affect instrument reliability and accuracy.

A licensing analysis related to ART research activity regarding I&C systems is provided in Table 5. A recommendation in this area is provided in Subsection 4.4. Implementation of Recommendation 8 in the near-term is expected to assist in:

- Development of capabilities for in-core monitoring and surveillance diagnostics of key parameters (power, flow, etc.) in advanced reactor environments to reduce the inherent uncertainties and resulting licensing challenges that result from the use of ex-core detectors or other less-accurate methods
- Development and refinement of methods for monitoring the performance of passive cooling systems, which are the key heat removal safety systems for most advanced reactor technologies
- Establishment of techniques and methods for inspecting and verifying the integrity of reactor internals.

2.1.7 Safeguards and Security

The NRC encourages designers of new reactors to integrate security into the design and conduct a thorough assessment to evaluate the actual levels of protection provided by those measures. Research may be necessary to assess the effectiveness of newly proposed security measures. Likewise, R&D may be needed to establish material control and accountability (MC&A) safeguards provisions that accompany new reactor technology and develop the technical basis that address current NRC criteria.

No licensing analysis has yet been performed related to ART research concerning advanced reactor safeguards and security. However, Table 6 has been reserved for future use. No licensing recommendations have been developed on this topic.

2.1.8 Accident Sequences and Initiators

Considering the range of scenarios and phenomenology that might be associated with a safety evaluation of a new design concept, extensive R&D may be necessary to ensure the analytical codes and models, along with the data used to support postulated scenarios and characterize operational phenomena, are available. This topic is closely related to the scope described under Subsection 2.1.3, "Analytical Codes and Methods".

No licensing analysis has been performed related to ART research concerning advanced reactor accident sequences and initiators. Research plans on this topic will require detailed interactions with design vendors to better understand how issues are to be addressed under their proposed design approach.

Table 7 is reserved for future use. A recommendation relating to establishing a venue for vendor interaction is presented in Subsection 4.4.

2.1.9 Probabilistic Risk Assessment

As noted by NRC,[1] reactor designs are to be risk-informed, thereby making the probabilistic risk assessment (PRA) process an important component in the overall reactor design process. Limitations associated with advanced reactor PRA experiences are anticipated with the application of system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems), to the risk metrics that are used (e.g., core damage frequency or large early release may not be the best figure of merit for some proposed reactor designs), to failure data, and perhaps most importantly, to the design, materials, systems, and safety approach. Both advanced reactor license applicants and NRC staff will need to determine the technical adequacy of the PRA and know if it is sufficient to justify the specific results and risk insights used in support of a license application.

The licensing analysis of current ART research regarding PRA activities is provided in Table 8. Future R&D on this topic is expected to benefit from a recommendation made in Subsection 4.4 that enables interaction with prospective applicants to provide better understanding how issues are to be defined and addressed under their licensing approaches.

2.1.10 Structural Analysis

Structural analysis tools for large LWR designs are mature and standardized, and benefit from an extensive applications database. It is unclear whether these tools can be widely used in non-LWR reactors without modification. However, confirmatory structural analysis of non-LWR technologies may indeed be possible to some degree with existing tools.

It is understood that additional research will likely be required regarding the qualification of seismic isolators. Some designs (such as the modular HTGR) are expected to use deep embedments many feet below grade which envelopes the reactor core and associated heat exchange systems. New seismic analysis tools may be necessary to support not only assessments of seismic impacts to below-grade safety SSCs but also to aid in developing the seismic isolation systems necessary to assure safety during a seismic event. Analysis tools for deeply embedded structures have not been used in a U.S. reactor licensing action. Advancements in structural analysis through ART-sponsored R&D could potentially crosscut multiple advanced reactor technologies and benefit a large portion of the regulated community.

No licensing analysis on the topic of structural analysis has yet been performed in relation to ART research priorities. Table 9 is reserved for future use. A research activity concerning seismic analysis of embedded structures has been analyzed in Table 2, Item 2.b., and is noted as a future licensing concern in Subsection 5.7.

2.1.11 Human Factors

Advanced reactors will present new operational and maintenance challenges that are different from current practices. The type and extent of these variations may affect areas such as the control room, use of computer-based technology as part of a digital I&C program, and call for modifications in alarms, controls, and displays associated with SSCs important to safety. Research considerations include definition of the functional requirements of the plant 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 the human factors topic. No licensing-related recommendations have been developed.

2.2 Quality Assurance

Advanced reactor applicants will be required to submit applications for a CP or an OL submitted in accordance with 10 CFR Part 50 or an ESP, a DC, or COL submitted in accordance with 10 CFR Part 52. These submissions must meet QA program requirements recognized by the NRC.

To support licensing, activities performed during technology development research and high-level design phases of the project should be conducted in accordance with applicable QA requirements already endorsed by NRC. These QA requirements must describe methods and establish applicable quality and administrative control requirements that meet 10 CFR, Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” This assures that activities supporting the regulatory safety review process provide adequate confidence that safety-related SSCs will perform their required safety function when required. These requirements may also be applied to certain equipment tests and research activities that affect non-safety-related SSCs yet support safe plant operations.

The QA program should be based upon American National Standards Institute (ANSI) N45.2, “Quality Assurance Program Requirements for Nuclear Power Plants,” and its’ daughter standards. The guidelines provided in ASME Standards 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)”, provide fundamental QA requirements for satisfying the 10 CFR, Part 50, Appendix B. Additionally, the following documents describe methods that the NRC staff considers acceptable for complying with the provisions of 10 CFR Part 50, Appendix B:

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

The general scope of QA in advance reactor deployment begins with technology development and high-level design activities and continues through final design, construction, and operation of the facility. Since future applicants will utilize R&D test data (and associated safety conclusions) when writing a license application, it is important to establish a sound QA program early during technology development and high-level design phases of the project; this includes research that may be done by ART. A quality assurance program description (QAPD) document is often warranted for ART development activities relating to advanced reactor safety. A QAPD (based on 10 CFR, Part 50, Appendix B) will provide and establish applicable QA requirements that meet the needs of NRC licensing.

Experiences gained during the NGNP project demonstrate the importance of establishing applicable QA requirements during technology development. For NGNP, a QAPD[3] was developed to address QA requirements established by 10 CFR, Part 50, Appendix B for executing work activities that generated technically-defensible scientific and engineering information to be used in future modular HTGR licensing applications. The NGNP QAPD was based NRC NQA-1-2008;1a-2009, “Quality Assurance Requirements for Nuclear Facility Applications”, as provided under RG 1.28. It should be noted that while the NGNP QAPD did address all 18 QA criteria established in 10 CFR, Part 50, Appendix B, not all were determined to be applicable by NRC staff during the technology development and high-level design phase of the NGNP project. In the case of NGNP, the QAPD will be reviewed and updated (or used as a reference) by future applicant(s) to establish QA requirements to facilitate later NGNP license application development.

As a point of current reader reference, the following QA requirements were deemed applicable to NGNP R&D and established in the QAPD during the technology development and high-level design phases of the project:

- Organization – Establishing the QA organization commensurate with the duties and responsibilities.
- Quality Assurance Program – Establishing the necessary measures to implement a QAP in order to ensure that the 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 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 procedure 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 the provisions of the QA will perform satisfactory 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 the established requirements.

These 14 reviewed elements of the NGNP QAPD were approved by NRC staff for later use in licensing.[4] The following QA requirements were deemed not applicable by NRC staff during the technology development and high-level design activities associated with NGNP and were therefore not endorsed:

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

The NRC staff also communicated their expectation to NGNP that either (1) a supplemented QAPD would be submitted should the scope of the NGNP be expanded to include design and/or construction activities in accordance with becoming an applicant under 10 CFR Part 52; or (2) any future applicant or licensee planning to design and/or construct a NGNP-type reactor based on NGNP research and development efforts would submit an independent QAPD covering the appropriate scope of activities in accordance with quality assurance regulations and guidance in place at that time.

The controls and quality program attributes discussed above should be considered and implemented as appropriate and applicable when performing the reactor technology development activities that are described in the remainder of this document.

3. ART REGULATORY TECHNOLOGY DEVELOPMENT PLAN

As a consequence of current ART priorities, RTDP analysis has thus far focused on research associated with two general types of advanced reactor technologies, i.e., the modular HTGR and the liquid-metal cooled SFR. A variety of DOE research planning documents related to these reactor concepts were reviewed to support this analysis.[5,6,7,8,9,10,11,12,13,14,15,16,17,18,19,20]

Additional information was collected through discussions with ART topical research area leads and subject matter experts to confirm research activity descriptions, status, and bring planning document information up-to-date where indicated. The activity information was then binned according to topical areas (discussed in Chapter 2) and entered into a tabular format for licensing impact evaluation. The evaluation was performed by ART Licensing staff and based heavily upon prelicensing experiences gained during the NGNP project. The analysis also sought to highlight long-term research activities essential to advance reactor technology licensing. Results of the analyses are documented in the tables presented later in this section.

The following subsections outline general ranking guidance and primary evaluation criteria used during the licensing impact analysis.

3.1 Regulatory Importance

Once a research opportunity is described, a statement is developed addressing how results of that activity (i.e., the data generated upon test completion) might support development of the technology safety case and support regulatory decisions about safety. The "Regulatory Importance" of a research activity focuses heavily on the role the activity is expected to play in supporting a positive outcome from the NRC safety review. A general ranking of regulatory importance is assigned to the activity such that:

- *High* – Phenomena or topic is of first order (fundamental) importance to design safety and is a critical component to the independent safety review process. Research information generated by the activity is understood as essential to successfully meet safety review criteria.
- *Medium* – Phenomena or topic is of secondary (contributing) importance to design, safety case, and the independent safety review process. Alternative regulatory options may be available to address the issue. However, the issue is made more important due to other factors, such as addressing concerns important to multiple advanced reactor technologies or considerations that significantly impact a timeline for deployment. While research activities with this level of regulatory importance are generally recognized as less imperative than items with "High" regulatory importance, completion of the research activity is still considered essential.
- *Low* – Phenomena or topic is not currently considered of significance to the design safety case or essential to support the independent safety review process.

These rankings are generally assigned based on consensus opinions of ART Licensing personnel.

3.2 State of Knowledge

Establishing a licensing priority recommendation for DOE research must also consider the current state of knowledge concerning the topic and the effect of "gaps" on essential information. Technology development research priorities should trend higher when inadequate knowledge exists to address and resolve a known safety issue. Guidance used when ranking the "State of Knowledge" criteria for an individual research activity consists of:

- *High* - A physics-based or correlation-based modeling capability is available that adequately represents the phenomenon or issue over parameters of interest. A body of data that would likely satisfy regulatory quality requirements exists to adequately validate and predict capabilities and/or support further development to completion (which includes NRC review and acceptance).
- *Medium* - A candidate model or appropriate means of correlation has been identified and is available to address most of the phenomenon or core issues over a considerable portion of the parameter envelope. Supporting data are available but the database is not necessarily complete or is of questionable quality, thereby allowing only a moderately reliable assessment of system capability.
- *Low* - No functional model exists or predictive capabilities are uncertain or speculative. No database exists and safety assessments cannot be reliably made due to high uncertainty with existing data.

These rankings are generally assigned based on information gathered during review of various ART research plans and the informed opinions communicated to ART Licensing staff by technical research area leaders and subject matter experts working within ART.

3.3 Status of Research

To further frame the licensing priority of individual research activities, an understanding is required concerning the state of knowledge development. Key informational gaps already identified and planned for resolution (or where research efforts are well underway to close data gaps in the foreseeable future) are to be noted. Additionally, once the status of a research activity supporting the safety case item is characterized, a time-phased work sequence should then be considered to ascertain if any major predecessor/successor relationships might exist between individual research activities and how those sequences might broadly affect a licensing timeline. Research status becomes particularly significant for activities with very long lead-times, that are sequentially dependent on the completion of other research, or both (as might be experienced by completing in-core fuels tests and post-irradiation examination (PIE), which in-turn generate data for subsequent use in MST analysis code development).

Research activity status is not discretely ranked but rather is focused on capturing information pertinent to time-related and/or data-needs issues. Questions related to these issues typically include:

- Is essential research already planned, underway, and resourced? If so, licensing priority can be reduced in recognition of a pending resolution to the issue.
- Do predecessor/successor relationships exist that affect research planning and performance? If so, licensing priority can be increased due to the influence sequential test plan successes may exert on the licensing critical path.
- Does the research address an essential concern in establishing or assessing the safety case? Do additional options exist to address the need? Licensing priority can be reduced if completion of the test plan is not considered essential to establishing the plant safety basis.

- Are suitable testing capabilities and essential infrastructure support available to support the investigation? If not, what are the precursors to launching the study? Licensing priority may be increased if testing infrastructure is unavailable to support essential research.
- Is there a long lead-time (i.e., >5 years) associated with the activity? Licensing priority can be reduced if a test plan or research activity 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 this information was done by ART Licensing personnel and tempered by recent regulatory perspectives and experiences. The result is a simple yet flexible preliminarily assessment scheme that projects expected licensing implications on discrete units of ART research. By capturing summary statements on anticipated regulatory importance, state of knowledge, and status of ART research activity, licensing evaluations and priorities can be derived which address potential ART research regardless of whether active research on the topic is already planned and/or underway.

A four-increment "licensing priority" structure was established to convey summary results of the RTDP analysis. Guidance on increment ranking values consists of:

High – A priority that indicates research activity results are expected to address a major safety case concern of great importance during licensing. Research activities with this priority rating generally exhibit a high or medium level of regulatory importance, the state of current technical knowledge would be low to medium with respect to information necessary to support safety case development and NRC safety decisions, and a long lead time can be expected for validated result generation.

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

Medium – This priority denotes research that tends to have a high or medium level of regulatory importance, the state of knowledge needed to support safety case development and the independent safety review process ranges from low to medium, but completion of the research plan is expected not to require a long lead time relative to safety case development. This rating may include lengthy research activities that are very important to regulatory safety decisions but are already well-planned and resourced. Research activities with this rating are acceptably scheduled according to currently understood licensing timelines.

A "Medium" priority denotes R&D activities on topics that are significant to the plant safety case and safety review process but are thought to present minimal risk to a licensing schedule. Mitigating factors in this priority class include short lead times for test plan completion or a need for additional design-specific information from designers to support proper test planning. A "Medium" priority can also identify research activities that are on a "watch list" for becoming a higher licensing concern in the future.

Low – This priority typically characterizes research activities with medium or low levels of regulatory importance, the state of knowledge can range from low to high with respect to anticipated data needed to support the regulatory process, 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 are expected to exert a low-level of influence in safety decisions. "Low" priority research items are not expected to challenge the overall critical path for license application development and the independent safety review process.

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

An activity with a "None" priority is often not considered an immediate ART research concern and is therefore outside the nominal scope of the RTDP.

While the aforementioned approach in establishing a licensing priority rank sets forth high-level guidance when evaluating technology-enabling research, it should be noted that the approach is not a rigid evaluative metric. Instead, the flexibility of the process allows for further priority adjustments in response to additional factors that may not be otherwise discussed and accounted for in Subsections 3.1 thorough 3.3.

3.5 ART Research Activities

Tables 1 through 10 identify and assess ART research opportunities that are of significance to advanced reactor technology development and commercial deployment. These tables communicate specific issues, factors and concerns relative to ART research that are pertinent to establishing the plant safety case and enabling an independent safety evaluation process. Information in each table is partitioned according to reactor technology (i.e., HTGR or SFR). Note that some tables are reserved for future use and will be completed as ART research activities potentially expand to address those topics.

Section 4 provides a summary of conclusions and recommendations derived from analysis of information contained in Tables 1 through 10. Section 5 identifies additional areas of R&D opportunity that might become a future licensing concern.

Description

Fuel Qualification:

Depending on specific attributes of the technology being considered, information relating to the design, manufacture, and use of fuel in support of a regulatory safety review may be a major concern for ART when identifying and scheduling R&D. A fuel qualification (FQ) program may require lengthy irradiation tests and examinations to evaluate new or modified fuels different than that currently used in LWRs. Because very long-lead times and critical infrastructure support is typically necessary to support comprehensive in-core irradiation tests and follow-on examinations, the nature and scope of outstanding fuels research need is almost always a priority licensing concern. Qualification of new fuel is an activity that demands extensive long-term planning that minimizes repetitive testing protocols and reduces uncertainty in resulting data.

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

Mechanistic Source Term:

A mechanistic approach to source terms development is necessary when establishing the technical basis for subsequent safety analysis and allocating appropriate credit for the radionuclide retention capabilities of a design. The advanced reactor safety design approaches that are proposed should be consistent with the presence of multiple barriers in radionuclide transport to the environment. Multiple barriers in radionuclide release are a basic expectation of the regulatory safety review process. A MST evaluation is based on detailed analysis of fuel and reactor behavior during normal operations and bounding accident scenarios. Source terms developed with a mechanistic approach must also identify and characterize radionuclide inventories that exist elsewhere in the facility. Mechanistic source terms 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 closely related to and reliant upon FQ research. The performance data associated with fuel type and core design provide the foundation for performing analytical MST modeling. Therefore, the licensing 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 operations and accidents**								
1.a	HTGR	Develop fuel service conditions for normal operations (supplemented by peak fuel temperature, burnup, fluence, burnup from fissions of bred plutonium, maximum time-at-temperature) and accident conditions.[a]	High	This research generates essential information that interfaces with LBE selection and accident analysis predictions. It is associated with fuel qualification and mechanistic source terms development.	High	Fuel service conditions are currently being addressed by Advanced Gas Reactor (AGR) Fuel Program irradiation tests which presumes a prismatic block core (pebble-bed core design is not currently a focus of AGR tests). Normal conditions are based on best available conservative code predictions for the fuel. Accident conditions are derived from best available information on the nuclear, thermal, and chemical environments predicted during anticipated LBEs for a preliminary NGNP design.	Test regimes addressing this issue are currently underway at INL.[b] Post irradiation safety tests on AGR-1 test fuel provided sufficient laboratory failure rate data to support initial conclusions about fuel accident performance. Questions still remain concerning how laboratory-scale data can be scaled up to represent industry produced fuel.[h] Multiple AGR tests are ongoing and planned. AGR test results may require further verification once fuel service conditions are defined through final design decisions.	Medium	Defining fuel conditions for normal and accident conditions is a key concern being addressed through the AGR tests. Issues dealing with research data scale-up will be part of design certification and should be addressed during license application development. Although task completion is important, licensing priority is reduced because AGR test program is proceeding on schedule. Verification of AGR results against final design conditions is a future priority task.

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, etc.), might affect fuel operating and accident performance.[a]	High	This activity factors into regulatory and technical assessments of adequacy concerning reliance on accelerated AGR test irradiations at the INL Advanced Test Reactor (ATR) to address higher ranges of fuel operating temperature, burnup, and fluence.	Medium	DOE/INL has yet to see evidence of significant parameter path dependence for normal fuel operating conditions. However, contingent upon on estimates of HTGR core design (yet to be finalized), further evaluation of this issue has been recognized by NRC as possibly necessary.[a]	Informational gaps in "path dependence" coverage are being taken into consideration in the planning of future AGR tests (5/6/7).[h] Additional irradiations dealing with fission product transport code validation (i.e., AGR-8) is delayed pending design information required to refine experimental planning. When completed, data from AGR tests will be evaluated for their significance using validated phenomenological models of TRISO fuel performance under operating conditions and accident conditions.	Medium	Path dependence variance is a NRC staff concern that was documented during NGNP prelicensing interactions. Evaluation of fuel operating parameter path dependence is an indicator of robustness in AGR test results; this concern is already planned for resolution. Future assessment will be needed to assure collected AGR data adequately represents the final design.
1.c	HTGR	Identify substantial uncertainties and undetected anomalies in fuel normal operating service conditions that involve key parameters such as maximum fuel normal operating temperature.[a]	Medium	Timely research attention will lead to clarifying critical needs and specific circumstances surrounding development and qualification of advanced sensor systems for HTGR prototype monitoring, surveillance, and testing. Inability to resolve these uncertainties could result in unnecessary conditions in the HTGR COL.	Medium	During prelicensing interactions, NRC staff documented the understanding that addressing uncertainties will likely require verification of initial and subsequent normal fuel operating conditions. This could be performed through special operational monitoring, testing, surveillance, and inspection programs at the (first) demonstration reactor.[a]	DOE/INL believes that planned AGR Irradiated Fuel Test #7 will demonstrate sufficient margins to failure for the TRISO fuel form under normal operating and potential accident conditions.[b] It is expected that test will experimentally address the uncertainty issue.[h] However, NRC staff currently believes HTGR core analysis and core monitoring issues can only be partially addressed by analytical means and that separate effects validation tests will be needed.[a]	Medium	A precise description and thorough understanding of in-core monitoring and initial power ascension tests must be established in conjunction with licensing the initial reactor module. Specific conditions remained to be defined and accepted by the NRC; these conditions should incorporate AGR test results as much as possible. Commitments on resolving remaining NRC concerns should be established early in the license application development process.
1.d	SFR	Evaluate SFR fuel acceptance criteria for normal operations and postulated accidents (consider core disruptive damage functions, cladding thermal creep strain limit etc.). Identify sources of significant uncertainty (e.g., burnup, fluence, thermo-physical properties) and how they may influence key parameters of interest (e.g., fuel and cladding temperatures).[c]	High	Understanding fuel behaviors and the parameters which influence fuel performance during steady-state irradiation and transient conditions (including anticipated operational occurrences and postulated accidents) is important in FQ and the selection and analysis of LBE. Understanding modes of fuel failure is a critical component in MST assessments.	Medium	Sufficient data and information from historic Experimental Breeder Reactor II (EBR-II) and Fast Flux Test Facility (FFTF) operations are likely available to support most SFR fuel designs provided they remain within the existing experience base of either metallic or oxide fuel.[k] These fuel data includes up to 10% burnup, peak cladding temperature of 600 C or less, peak dpa of 100, and use of un-reprocessed fuel.[c] Acceptable margins in this experience base include up to 20% burnup, 650 C peak cladding temperature, and variations in fuel pin dimensions.[k] Confirming the adequacy of this state of knowledge requires design-specific information.	SFR metal fuel irradiation testing and physics analysis databases are being developed at Argonne National Laboratory (ANL) under DOE-NE's ART program.[c] Although fuel acceptance criteria (to be defined by the design vendor) is expected to remain nominally within boundaries of the existing database, some deviations (i.e., use of advanced alloys for cladding material, different fuel pin dimensions, and even higher burnup) can be expected.[k] The importance of these variations still needs to be assessed.	High	Interactions with SFR designer authorities and NRC staff are needed to assure essential fuel qualification gaps are identified and resolutions appropriately planned to complete the existing experience base. Additional fast reactor fuels-related testing could be a very complex, long-lead time activity. The results of gap analysis will need discussion with and confirmation by NRC staff.

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	Assess data quality levels and data configuration control standards associated with SFR metal fuel irradiation testing and physics analysis databases. These databases are being developed under DOE-NE's ART program to support future licensing activities.	High	Extensive legacy data will be used by applicants and NRC to perform a plant design safety analysis. Early regulatory interactions concerning assessment of heritage data quality and the configuration standards necessary to support FQ are essential to allow use of data from EBR-II and FFTF concerning operations, irradiation experiments, and safety tests.	Low	Domestic SFR fuel knowledge is predicated mostly on historic EBR-II and FFTF experiences. Confidently assessing the state of knowledge requires additional design-specific information and confirmation from regulators that these data are acceptable for use in licensing.[g]	SFR metal fuel irradiation testing and physics analysis databases are being developed at ANL under DOE-NE's ART program. Data quality levels and configuration control standards applicable to these databases remain to be assessed and implemented with respect to satisfying licensing-related criteria.	High	Assessment of EBR-II and FFTF information and comparison with yet to be determined design approach data needs are a high priority precursor to identifying analytical data gaps and planning subsequent fuel research. Additional fuels research is likely a complex, long lead activity. Acceptance of existing data for safety review use must be confirmed with NRC.
2	**Demonstrate fuel performance requirements at normal operating conditions are met using irradiated fuel at design conditions, fuel irradiation performance monitoring, and post-irradiation examinations**								
2.a	HTGR	Perform irradiation, safety testing, and PIE of both UCO and UO_2 TRISO fuel from laboratory and prototypic scale equipment to obtain normal operation conditions performance data.[b]	High	This research broadens options and enhances prospects for meeting TRISO fuel performance requirements. It supports a fundamental understanding of relationships between the fuel fabrication process, as-fabricated fuel properties, normal operation, and associated accident condition performance.[b]	Medium	When completed, multiple AGR test results will provide necessary irradiated fuel performance data and provide irradiated fuel samples for safety testing and PIE concerning key fuel product and process variants.[b]	With AGR Test 2, the laboratory scale AGR irradiation test phase is already concluded; PIE is underway. Prototypic scale testing is still necessary and planned in conjunction with performance of AGR Tests 5/6.[h]	Medium	This research is an important licensing concern but already has a good state of knowledge. Completion of AGR Test 2 PIE and the prototype testing of AGR Test 5/6 will confirm information necessary for development of a license application. Priority is reduced because necessary research is either underway or planned and tracking towards completion.
2.b	HTGR	Perform irradiation, safety testing and PIE of representative fuel containing design to fail (DTF) particles in support of fission product transport model development.[b]	High	The AGR test program allows assessment of the effect of impurities on intact and DTF fuel performance and subsequent fission product transport. This information is essential to MST development. AGR tests will also provide irradiated fuel performance data on fission product gas release from failed particles and irradiated fuel samples for safety testing and PIE.[b]	Medium	The in-pile gas release, PIE, and safety testing data on fission gas and metal releases from fuel kernels will be used in the development and refinement of improved fission product transport models.	Necessary AGR irradiation tests to acquire these data are completed and PIE is scheduled. Research activity is scheduled and on track for completion.[h]	Medium	Completion of this activity is necessary to design safety analysis. Topic already has good state of knowledge. AGR test results are being collected to provide necessary confirmation. Priority is reduced because research is well underway and scheduled for completion without adversely affecting deployment critical path.

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 irradiation testing.[a]	Medium	The lack of TRISO fuel performance data obtained in a (real-time) modular HTGR neutron environment has been documented as a concern of NRC staff.[a] Fully prototypic data may be required to demonstrate and confirm that fuel performance is adequately understood and can be predictively modeled with respect to fuel radionuclide retention and transport.	High	AGR-related PIE and safety testing intend to provide a broad range of data on fuel performance and fission product transport within fuel particles, compacts, and graphite materials representative of fuel element blocks. These data, in combination with in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios) will be used to demonstrate compliance with fuel performance requirements and support the development and validation of computer codes.[b]	Multiple AGR irradiation tests were designed to provide necessary data and sample materials to support licensing.[b] Based on existing AGR test plans and status of research performance, further research on this issue is not considered an obstacle in demonstrating required fuel characterization and performance.[h]	Low	AGR tests are designed and performed to represent fuel safety using accelerated test conditions. While comparisons may later be performed to formulate additional conclusions, additional research on this issue is not a significant concern.
2.d	HTGR	Evaluate plutonium generation and burnup in fuel test irradiations.[a]	Low	Plutonium burnup is among the normal operating service condition parameters to be specified for HTGR fuel. While DOE/INL currently believes this issue has little effect on HTGR TRISO fuel performance, regulators have documented this issue as an ongoing concern during prelicensing discussions.[a]	Medium	The need for further research on this topic does not appear relevant to the pebble bed HTGR design.[h] For prismatic designs, DOE/INL's current approach is to increase plutonium burnup in the 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 will provide necessary data and sample materials to further existing knowledge on this issue. A test program is under the VHTR TDO Fuel Development and Qualification program.[b]	Low	NRC has indicated they want to thoroughly understand plutonium generation parameters in HTGR fuel.[a] AGR tests are already planned which are anticipated to address the issue adequately in prismatic HTGRs. Discussions with NRC on this topic should resume once AGR test data becomes available.
2.e	SFR	Ensure existing SFR fuels irradiation testing and PIE are available as required to support fuel design. If data is required that is significantly outside the existing experience base (i.e., with substantial deviations in pin dimensions, fuel compositions, and higher burnup), perform additional testing.	High	It is currently presumed that sufficient historic data exists to support a regulatory safety review of SFR fuel designs.[k] However, additional fuels testing research may be necessary to broaden design options, increase assurance that fuel design performance requirements are met, and understand relationships between the fuel fabrication process, resulting fuel properties, and fuel performance under 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 that scope 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 vendors.[k]	Knowledge preservation efforts regarding SFR fuel under DOE-NE's ART program include EBR-II Metal Fuel Irradiation Test Database and Physics Analysis Database. An effort to develop an FFTF Fuels Irradiation Test Database is yet to be initiated. Sufficiency of existing data for licensing purposes is possible but must be confirmed.[k] Should existing data contain ambiguous and/or incomplete information for a regulatory safety assessment, additional SFR fuels testing will be necessary; this will be a complex, long lead time activity if required.	High	Understanding and predicting fuel performance during all design conditions is a key licensing issue. Based on current understandings, sufficient experience bases may exist from past SFR operations. However, questions concerning data quality, extent of data coverage with respect to emerging designs, and the data needs of the regulatory safety evaluation process remain unanswered. Until the need for additional fuel testing is established, this activity is considered a high 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
3	**Demonstrate fuel performance requirements for accident conditions are met using statistically significant irradiated fuel at accident conditions and monitoring fuel accident performance**								
3.a	HTGR	Demonstrate the scope of fuel performance testing for LBE accident conditions. Ensure conditions like reactivity excursion events, moisture-ingress events, and air-ingress events are adequately understood and factored into fuel performance requirements.[a]	High	Results of topical research are important to interface LBE selection and associated accident analysis predictions effectively with FQ and MST development.	Medium	AGR PIE and safety testing is planned to provide a broad range of data on fuel performance and fission product transport within fuel particles, compacts, and graphite materials representative of fuel element blocks. Additional data, coupled with in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios) are necessary to demonstrate compliance with fuel performance requirements and support the development and validation of computer codes.[b]	Fuel performance in connection with moisture and air ingress events will be characterized at the conclusion of AGR Test 5/6/7 PIE. Existing reactivity excursion data are currently sufficient to support design and licensing; no plans exist for additional reactivity testing.[h]	Medium	Understanding fuel performance during design accident conditions is an important issue. However, adequate data currently exists or is planned for generation in conjunction with forthcoming AGR tests. Priority is set recognizing R&D is already planned and scheduled to address this issue.
3.b.	HTGR	Perform irradiation testing, safety testing, and PIE of the qualification test fuel to demonstrate that reference fuel meets HTGR fuel performance requirements during accident conditions. Obtain data still needed for fuel performance model validation.[b]	High	Research provides fuel performance data and irradiated fuel samples for PIE and post-irradiation heating test/PIE in sufficient quantity to validate fuel performance codes and models and demonstrate capability of fuel to withstand expected conditions in support of plant design and licensing.[b]	Medium	When completed, AGR testing 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 accident conditions.[b]	Fuel qualification testing and examinations related to accident conditions are scheduled in conjunction with AGR Tests 5/6/7.[h] Data will be developed at that time, which can support fuel performance model evaluation.	Medium	Research activity has high regulatory importance and a medium state of knowledge. Acquisition of additional data and information is already planned through AGR tests. Licensing priority reflects the anticipated completion of this activity.
3.c	SFR	Conduct safety testing (if necessary) to ensure key fuel transient behavior and parameters that affect fuel failure modes are understood and factored into performance requirements. Ensure data can support transient fuel performance model validation, is reproducible, and bounds the physical phenomena that could degrade SFR fuel performance under off-normal and accident conditions.	High	Knowledge of transient fuel behavior is essential to effectively interface LBE selection and analysis with fuel qualification and mechanistic source term assessments. Physical phenomena that could degrade SFR fuel and contribute to radiological source terms must be understood under all anticipated off-normal and postulated accident conditions to predict fuel performance and assess consequences of fuel failures. Fuel performance data and PIE of irradiated fuel samples are also important for validation of fuel performance models.	Medium	Data from past safety testing and PIE does exist to demonstrate fuel performance during a wide range of postulated accident conditions. Existing characterizations of medium range burnup (<10 %) fuel may prove sufficient for licensing an SFR plant under normal operational conditions but is contingent on the type of fuel chosen through design decision and data requirements of the regulator.[d] Experiments have been performed concerning fuel movement and transport during transient overpower conditions. Gaps for irradiated fuel beyond 10% and novel fuel design concepts (such as vented fuel) may require additional testing.[k]	Knowledge preservation efforts regarding transient SFR fuel behavior under DOE-NE's ART program include EBR-II, FFTF, and TREAT safety testing databases. Evaluation of data sufficiency for licensing purposes has yet to be initiated. Additional transient SFR fuels testing will be necessary in some key areas such as fuel failure modes under loss of flow conditions. Transient fuel tests to address beyond design basis accidents is considered a long-lead activity that will require unique infrastructure testing capabilities that are currently very limited or unavailable.	High	Ongoing data recovery efforts under DOE-NE ART program are essential. Data gaps may still exist relative to fuel performance. Additional transient fuel tests may be required for some BDBE accident conditions. These tests (if required) are likely a long-lead activity with little infrastructure available to support fuels testing. The potential for major informational gaps on the topic makes it a high 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
4	**Establish and validate models for fuel performance and radionuclide transport in fuel**								
4.a	HTGR	Perform irradiation, safety testing and PIE of the qualification test fuel to support of fission product transport code validation.[b]	High	A validated fission product code that is reviewed and endorsed by the NRC for use in a safety analysis is essential when developing MSTs that support licensing.	Medium	A multi-monitored AGR test train is planned which will include fuel compacts seeded with fuel particles that are missing buffers and will be subjected to different temperatures among capsules. The test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to validate fission product transport codes.	Experiments associated with AGR Test 8 will be irradiated in the ATR flux trap housed in one test train or a Large B position.[h] Data from this test will be used to validate the transport code.	Medium	Research activity has great regulatory importance but medium knowledge. ATR Test 8 results are essential to resolve this concern but AGR Test 8 is not currently scheduled due to a lack of necessary plant design information. Once the design is sufficient to support AGR 8 test planning, the irradiation should be scheduled and performed.
4.b	HTGR	Resolve outstanding uncertainties regarding flux-accelerated diffusion of metallic fission products during irradiation.[a]	Low	This issue is not viewed to be of significant regulatory importance but rather is an issue about collecting confirmatory information about a topic DOE/INL believes to be already accurately characterized.	High	The intent of the AGR irradiation, post irradiation, and safety testing is to obtain data on FPT through fuel matrix and fuel element graphite with a known source of fission products in the fuel compact. This allows measurements and evaluation of the fission product gradient across the matrix and graphite surrounding the fuel compact through PIE.	Critical review and analysis of historic data on both in-pile and out-of-pile fission product diffusion in TRISO-coated particle fuel is underway. For the select AGR tests, DOE/INL will use PIE to measure the release of fission products under irradiation, analyze these measurements to establish diffusion coefficients under irradiation, and compare the resulting diffusion coefficients to the historic values from IAEA-TECDOC-978.	Low	Further research in this area is considered to have relatively low regulatory impact as the current state of technical knowledge is good.
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 the gap are conservatively neglected. The NRC staff views the DOE/INL approach as reasonable for use in the context of a conservative consequence analysis. Consequently, this issue has insignificant regulatory impact.	High	Calculation of event-specific MST for the prismatic core presumes the fuel compact-to-graphite gap to have no effect on the transport of gaseous fission products.	Further research on this issue has been incorporated into the AGR 3/4 fuel testing program. Conclusion of those tests and PIE will fulfill data needs.[h]	None	Because of the conservative presumptions already made in this topic with respect to safety, details associated with further research on this issue do not have a licensing impact.
4.d	HTGR	Develop transport models for all radiologically significant radionuclides in modular HTGRs.[a]	High	A robust capability to conservatively model and predict radionuclide transport from point of generation within the fuel to offsite receptors is an important element in MST development and when conducting associated safety reviews.	Medium	It is the DOE/INL position that collection of data on all radionuclide species that are analyzed in the calculation of mechanistic source terms is unnecessary. DOE/INL proposes to classify each radionuclide and species into one of nine radionuclide classes (established based on similarity of chemical and transport properties) and conducting analysis according to class properties.	DOE/INL is working on developing experimental data of fission product transport for representative classes of radionuclides (e.g., Cs-137 for alkali metals, I-131 for halogens) and applying that to model other radionuclides in that class.[b] The AGR test program has not scheduled extensive testing related to tritium transport.[h]	Medium	The proposed data application approach appears sound with respect to the needs of license application development. However, further review and approval is still required by NRC. HTGR outlet temperatures of 750°C do not create a strong concern in tritium transport but a gap will be created if the design outlet temperature increases significantly. Further interaction with NRC is warranted once 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 models for fuel particle performance during normal operation, heat-up accidents, and reactivity accidents that 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	It is essential to develop accurate and valid models that predict tristructural istotropic (TRISO) coating degradation and failure phenomena under normal, off-normal, and accident conditions. Experiments must adequately envelope all LBEs that involve air or moisture ingress present in the final design as these issues may affect particle failure fractions releases of iodine, metallic fission products, and fission gasses.	Medium	DOE/INL uses the 1989 Goodin-Nabielek model for fuel performance. Understanding important material properties is necessary for accurate modeling under 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 does depend on plant design and analysis which are yet to be established. DOE/INL research has concluded that oxidant contaminants will encounter extensive reactive material before reaching fuel particles despite relatively rapid oxidant diffusion through matrix materials.	The AGR program will develop fuel performance information of coated-particle fuel (either UO_2 or UCO) that are more 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 must 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	Fuel particle performance models for normal operations and LBEs are a major topic of regulatory interest. Necessary support research data is available (or soon will be available) and essential model development is underway.[a] Some additional research efforts may be necessary as a function of decisions yet to be made about design but those efforts are contingent upon the applicant.
4.f	SFR	Identify, describe and confirm all significant radionuclide transport phenomena and assumptions for SFR fuel. Develop fuel behavior models to predict the margin to cladding failure and contribution to source term during postulated beyond-design basis accidents.	High	A validated predictive capability for margin-to-cladding-failure assessments during postulated accidents is important for design review. The unique effects of fuel pin sorptivity and interaction with sodium coolant plays a major role on fission product transport in the SFR and must be appropriately characterized and quantified for safety assessments.[k]	Medium	Continued development of fuel behavior models during postulated accidents that could lead to fuel failures is being pursued under DOE-NE's ART program.[k] Experiments have been performed concerning fuel movement and transport data are available for validation of these models. Radionuclide release from metal fuel is well understood for cladding failure scenarios and low-burnup fuel melting. However, more mechanistic approaches for modeling radionuclide release into sodium from molten metal fuel at high burnup may be necessary.	Current ART efforts are focused on legacy data recovery and continued model development. An approach to address compliance of the databases to QA licensing standards is needed. A high-level survey of existing research SFR code capabilities has been conducted.[d] However, these codes were developed for use in research and may not have been subject to revision control. Continued improvement and validation of these codes will also depend on their regulatory acceptability.	High	Continued development of fuel behavior and radionuclide transport models based on more mechanistic approaches, and appropriate validation of these models, is a high priority.
5	**Develop fuel product and fuel fabrication process specifications**								
5.a	HTGR	Develop fuel design, fuel fabrication product specifications, and fuel fabrication process specifications that support the HTGR safety case.[a]	High	Due to the role TRISO-coated fuel plays in the source term, establishing and meeting HTGR fuel specifications is critical to a regulatory safety analysis that demonstrates the design satisfies top-level NRC requirements. These requirements are stated in terms of dose consequences for occupational exposures, siting, safety goals, and objective of doses below the EPA protective action guidelines at the site boundary for all LBEs.	Medium	AGR fuel qualification program includes requirements to develop fuel product and process specifications for large-scale TRISO fuel fabrication. This will define requirements the fuel must satisfy to ensure acceptable fuel performance under HTGR operating and accident conditions.[b]	Multiple elements of fuel design and manufacture remain to be finalized. The fuel specification has been established and will be validated in connection with AGR tests 5/6.[h]	Medium	Developing a TRISO-coated fuel specification is critical for future commercial fuel manufacturing. A fuel specification is currently available and AGR tests 5/6 (scheduled to start in early 2017) will confirm the integrity of TRISO fuel design. Licensing priority is medium in recognition of planned issue resolution.

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.b	HTGR	Develop and demonstrate a fuel fabrication process that equals or exceeds fuel fabrication requirements as determined by applicable source term calculations. The process is to include adequate margins of safety and address 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 fuel fabrication process set to proper specifications. Research demonstrations on high-reliability TRISO fuel fabrication processes are critical to verifying final fuel performance acceptance in terms of fuel particle failure rates and fuel radionuclide transport characteristics during normal operations and during accidents.	High	DOE/INL has developed a considerable body of technical information on the manufacture of TRISO-coated fuel that meets required TRISO-coated fuel particle failure rate specifications during normal operation and heat-up 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-inch prototypic production-sized coater.[b]	The TRISO fuel fabrication process and setting product specifications for fuel qualification tests are very close to finalization. The primary challenge remaining in this area is to "optimize" the fabrication process.[h]	Low	Necessary investigative research activities are nearing completion. Existing levels of knowledge are very good in support of fuel fabrication capabilities. Licensing priority is significantly lowered in recognition that required research is nearly complete.
5.c	SFR	Develop fuel design, fabrication, and process specifications to reliably produce fuel with requisite levels of quality. Include adequate margins of safety and consider factors that affect in-reactor fuel performance.[c]	Low	Establishing appropriate SFR fuel design specifications may be required as part of fuel qualification efforts to assure fabrication process adherence.	High	EBR-II and FFTF fuel design and fabrication experience is available to support this topic but it remains to be assessed for usability, quality, and comprehensiveness.[c,g] If applicants seek to use novel concepts such as vented fuel, new support fuel characterization and measurement methodologies will be required. For proposed designs that use advanced alloys as cladding materials (such as HT9M), additional design, fabrication, and process specifications will be needed.[k]	Criteria for assessing SFR fuel performance have been established using results from laboratory tests as well as the experiments in EBR-II, FFTF and the Transient Reactor Test (TREAT) Facility at INL. This heritage information is currently being recovered under DOE-NE's ART program. Gaps in key data remain to be assessed in conjunction with design criteria that are still emerging. Resolution of certain data gaps may indicate a long-lead research initiative is needed to develop necessary technology.[f] New measurement and characterization methods may be necessary that affect the fuel fabrication process.	Low	Accessing detailed fuel specification and fabrication records that already exist can facilitate or eliminate future fuels development needs. Should the concept of tolerating some fuel failure or venting fission products be sought by applicants, early prelicensing interactions with NRC will be necessary to ascertain the necessary research needed to support regulatory acceptance.
6	**Conduct irradiation and accident proof testing of fuel fabricated on the production lines of the fuel fabrication facility**								
6.a	HTGR	Conduct irradiation proof testing and post-irradiation heating of fuel that is produced in a TRISO fuel fabrication facility to demonstrate the acceptable performance and quality of the fuel.[a]	Low	Irradiation proof test and post-irradiation heating tests of fuel produced in the TRISO fuel fabrication facility is necessary to demonstrate acceptable performance of the fuel and qualify the fuel. These tests are required for HTGR technology and are expected to be performed by the fuel vendor at a later time. It is not essential for current plant R&D purposes.	Medium	DOE/INL will use mixed batches of fuel made on the single production-scale line for certain AGR tests to simulate the variability of fuel made on the fuel fabrication facility lines for the prototype.[a]	This activity is not currently included in the AGR Fuel Program. Proof test by a fuel vendor would rely on data generated by PIE and post-irradiation heating tests generated by AGR testing. Vendor tests would be expected to be largely confirmatory of precursor AGR test data.	None	Manufacturing tests are expected to be confirmatory of precursor AGR test data. No ART R&D attention is currently directed towards this issue.[h] Future applicants must consider and address this activity.

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
6.b	SFR	Determine if production line proof tests of fabricated SFR fuel are necessary as a function of associated potential to affect and contribute to mechanistic source term and plant safety.	Low	Demonstrating that fabrication specifications for SFR fuel are met is an issue that fuel vendors are expected to address. The extent and surety of data necessary for the demonstration will be primarily driven by anticipated fuel performance expectations.	High	Although no SFR fuels are currently being made domestically, there is a historical record concerning fabricated SFR 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 recovery of relevant data.	Low	Research on the topic is not yet recognized as a priority for SFR fuels types for which substantial experience exists.
7	**Develop event-specific mechanistic source terms**								
7.a	HTGR	Calculate a MST for specific HTGR 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 demonstrated as plausible and conservative is critical to evaluating plant safety and in establishing the site boundary and emergency planning zone. 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, the issue has become largely a regulatory concern rather than a topic for future research.	High	Modular HTGR precedents and AGR test results have already established a sound technical basis for source terms development. NGNP regulatory white papers and public meetings with NRC have conveyed NGNP positions to NRC staff on how MST should be calculated and used in siting decisions.[e] While proposed approaches were generally found reasonable by NRC staff, final acceptance of these approaches remain to be finalized by the applicant in future COL application-related actions.[a]	Additional data pertaining to MST will be collected through AGR Tests 3/4 (now underway).[h] When completed, the primary hurdle existing in issue resolution is associated with regulatory interactions between NRC and the applicant. Future license applicants should develop detailed licensing plans and resume regulatory discussions with NRC staff early to finalize already proposed NGNP approaches.[e]	Low	Adequate technical information is available or will be available soon through AGR test plan completion that adequately supports development of the modular HTGR MST. Unaddressed gaps on this issue are not currently significant with respect to research and must be addressed by future applicants.
7.b	SFR	Develop plausible MST models for specific bounding SFR LBEs to demonstrate compliance with the regulatory requirements and safety expectations conveyed in the Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants.	High	A conservative MST that characterizes radiological releases for all operational modes and postulated accidents is critical to SFR plant safety and defining necessary site boundary and emergency planning zones. Without a source term, a plant regulatory safety analysis cannot be completed and a license will not be issued. Given the unique nature of certain SFR MST elements, early interactions with NRC staff will be necessary to ensure MST development approaches are acceptable for NRC reviews.	Low	A technical basis for an SFR MST exists based on historic EBR-II and FFTF data, extensive past experimentation and metal fuel accidents. The metallic fuel MST information has recently been qualitatively characterized and linked to emerging SFR designs.[i] However, a design concept that calls for vented fuel may require new support information. Because precedents do not exist for such MST elements, early NRC interactions regarding data requirements and qualification, supported by relevant test data (which may not currently exist) will be necessary.	ANL has characterized the history and major (qualitative) components of the anticipated SFR MST (assuming a metal-alloy fuel pool-type design).[i] The research required to quantify these components, particularly those dealing with accident conditions, remain to be defined, planned and performed to enable conservative predictions of radiological transport and release under the major LBE scenarios. Existing data may be sufficient to support a complete MST model as would be required for licensing purposes but is dependent upon design and accident specifics and regulatory qualification requirements.[i]	High	Examination of MST for LBEs is an essential step in license applications. Operational information and data from safety testing is available and some data are available from past accidents. However, new design features not represented in past operations must be captured and characterized. Given that a detailed MST characterization has not been done for a SFR, the MST model should be developed and interactions conducted with NRC staff to further clarify the 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
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 impact exerted by reactor building vent/filtration system on mechanistic source terms.[a]	High	Understanding behaviors and having the ability to predicted transport behavior of radionuclides in the HTGR primary circuit and reactor building are essential to developing a plausible mechanistic source term. Modeling capabilities of these behaviors is required during regulatory safety reviews to satisfy siting and design goals.	Medium	Correlations for predicting radionuclide re-entrainment during primary circuit depressurization transients have large uncertainties and are not yet adequately validated to support predictions. Historic data is not extensive for HTGRs on this topic and display large scatter. Studies have been conducted to assess design options for the reactor building and the advantages/disadvantages of each option.	The AGR Fuel Program does not currently 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 specific design information will be necessary from the applicant to support research planning.[h] When test planning is enabled, 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 is critical a aspect of radiological safety in plant design. Predictive modeling capabilities of these behaviors are required during regulatory safety reviews in order to satisfy siting criteria and design goals. Additional specific design information will be necessary from the applicant to support further research planning.
8.b	SFR	Determine radionuclide transport behavior in the SFR primary coolant system and containment structure to support mechanistic source term predictions during the postulated beyond-design basis accidents.[c]	High	Understanding radionuclide transport behavior in the primary circuit and SFR containment building is critical to developing a more comprehensive mechanistic source term model that subsequently enables impact evaluations at offsite receptors.	Low	The unique role liquid metallic sodium will play in radionuclide transport suggests considerable data will be needed to support code development. SFR technology experts indicate that the effect sodium plays in radionuclide transport is not well characterized (especially during accident events) and is a topic for future research.[l] Continued development in capability to model SFR primary coolant system and containment response for MST assessments is an ongoing effort under DOE-NE's ART program.	There is substantial past experimentation and several accidents that provide insight into radionuclide release from metal fuel (during pin breach and with fuel melting at low burnup) and transport in the primary system. Additional testing of radionuclide release from high burnup molten metal fuel may be necessary. Research requiring only radionuclide tracers (e.g., radionuclide release from fuel debris into a quiescent sodium pool and radionuclide behavior in containment), could be conducted using existing facilities.[k]	High	Understanding radionuclide transport behavior in the primary circuit and containment building is critical to developing a comprehensive SFR MST model. Uncertainties remain on the topic and additional research data is expected as necessary to address gaps. Pending a path forward to address the gaps, the topic is assigned a high regulatory concern.
9	**Demonstrate mechanistic source term models in best estimate and conservative analyses of transients and accidents**								
9.a	HTGR	Establish an evaluation methodology for addressing HTGR MST uncertainty and determine associated comprehensiveness. Include a basis for the terms "best estimate" and "conservative."[a]	High	Mechanistic source terms models must show reasonable degrees of comprehensiveness and certainty to justify their use in siting and design basis decisions. NRC must review and endorse proposed approaches when they are used for purposes of licensing.	Medium	Approach for accident consequence analysis relies on calculation of event-specific mechanistic building-release source term and associated dose rates, which is 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 domain of active research planning.	The AGR fuel qualification program will generate data that can be used to confirm that mechanistic source terms under normal and accident conditions are accurate to within prescribed limits.[b] However, it is currently understood that the likely applicants for an HTGR license do not have a recognized capability to quantitatively develop such a methodology.[h]	Medium	Development of a methodology for addressing MST uncertainties is critical for plant siting and for characterizing the safety design basis. NRC must review and endorse this methodology when it is developed. INL has the data and capability to develop this methodology but the activity is not within current work scope.

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.b	HTGR	Develop mechanistic source terms for specific HTGR LBE categories.[a]	High	Development of licensing MSTs is a function of the LBE categories as proposed by an applicant. The issue is critical to a design basis evaluation and siting analyses. Although extensive prelicensing interactions have occurred with NRC staff concerning MST development approaches, similar interactions regarding HTGR LBE category development has been inconclusive.[e] The MST must be related to the specific LBE categories proposed by the applicant and represented in final elements of design.	Medium	Upon completion of planned AGR tests, the major elements of the HTGR mechanistic source terms addressable by technology research and development 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 the models already proposed to NRC. The HTGR's most important barrier to fission product release (i.e., the coated fuel particles) will be modeled on a statistical basis to account for uncertainties about a mean in the particle failure probability. However, linking source terms to specific LBE categories will require specific design approach information from future applicant(s); this information is not currently available.	Medium	Development of MST is a function of the LBE categories as proposed by an applicant. Extensive prelicensing interactions have already occurred with NRC staff concerning MST development approaches. Definition of LBE categories remains a matter of considerable uncertainty and requires involvement of the applicant. Further interactions with the NRC staff should resume when applicant is identified.
9.c	HTGR	Obtain peer review of mechanistic source terms.[a]	Low	Peer review is a standard part of the PRA development process and need not be a dedicated HTGR research issue.	High	Peer review of the PRA is standard in the nuclear industry. The process is well understood and available for use.	PRA elements will be peer reviewed, including source term calculations.	None	Peer review process is not seen as a significant ART research concern.
10	**Develop prototype pre-operational and operational programs to verify and supplement the developmental technical bases for fuel qualification and mechanistic source terms**								
10.a	HTGR	Evaluate application of prototype provisions of NRC regulations to facilitate initial 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	The potential for undetected anomalous or off-normal operating conditions may require consideration when establishing initial plant operating limits. The presence of unknowns is a factor in both long-term and immediate pre-accident operating histories that are used in licensing safety analysis. Conclusions of the analysis may require supplemental confirmation through prototype tests, surveillances, monitoring, and inspections.	Medium	The purpose of prototype-specific design features and programs is to verify that initial and evolving operating conditions and performance elements (e.g., fuel performance) developed based on research-level results are consistent with those predicted and considered as the technical basis for licensing.	No ART research is currently planned in this area. DOE/INL can advise how design features, testing, and surveillance programs specific to the HTGR demonstration plant can be addressed and used to verify and supplement developmental technical basis now being established for HTGR FQ and MST.	High	Understanding the requirements and resolution of associated prototype plant issues will require involvement of the applicant. If prototype provisions are employed as an option in HTGR licensing, interactions with NRC should be initiated by the applicant and can be expected to result in additional conditions imposed on the initial facility. This item remains a major licensing concern.
10.b	HTGR	Determine the prospective challenges and possible need for physical verification of normal fuel operating conditions in HTGR cores.[a]	Medium	Accident source terms in modular HTGRs are sensitive to core operating conditions. Inherent technical challenges in monitoring HTGR cores during normal operating conditions make measurements difficult to perform. Should in-core measurements be required as a condition of initial module licensing, instrumentation reliability may be a significant challenge.	Medium	Multiple factors are known to contribute to difficulties in predicting normal operating conditions in prismatic-block and pebble-bed HTGR cores. In both pebble bed and prismatic reactors, the typical operating temperatures are too high for most thermocouples. Additional thermocouple development would overcome this limitation. However, in pebble bed reactors, instruments cannot easily be inserted into the core at all. Melt-wire pebbles could be dropped into the core to obtain data on peak coolant temperatures from which local fuel temperatures can be calculated. As it is difficult to precisely place and track these pebbles, uncertainties are inherent to the process.[j]	DOE/INL is capable to develop 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 a combination of some measurements with new thermocouples and better core simulation capabilities should be able to bound uncertainties in the core temperature profile.[j]	Medium	Current thermocouple technology does not fully enable HTGR in-core temperature monitoring at this time. However, with some additional R&D, approaches can be developed and implemented if NRC requires such monitoring. A needs determination for this monitoring will require applicant involvement and commitment. This issue is a possible future licensing concern that must be addressed later in conjunction with development of license application.

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 use of regulatory prototype provisions to facilitate prototype licensing approach for SFR technology. Items for consideration include verifying and supplementing developmental technical basis for fuel qualification and MST, SFR fuel service conditions, and fuel performance.	High	Without adequate supporting data and information about plant safety margins, licensing a commercial SFR may require deployment of a prototype plant so that uncertainties can be assessed and operating limits refined. However, prototype nuclear operations require larger safety margins and entail additional programs concerning measurement, testing, surveillance, monitoring and inspection. Establishing plant operations under prototype regulations is a complex and uncertain process that requires detailed early interactions with NRC staff.	Medium	Due to limitations in SFR licensing experience and scare infrastructure capable of supporting further testing in key areas like fuel qualification, a prototype approach in SFR technology appears to be a reasonable expectation for initial commercial deployment.[g]	Active ART research initiatives are not geared to support deployment of an SFR prototype plant. Detailed coordination with the prototype plant applicant will be necessary to support planning for this still unique regulatory option in reactor deployment.	High	Until a firm understanding of the role a prototypical plant licensing action will play in the strategy for commercial SFR licensing, this activity has a high level of regulatory concern and is a topic of potentially significant research interest.

a. NRC, "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms (Revision 1)", ML14074A845, Encl 2, July 17, 2014
b. INL, "Technical Program Plan for the Very High Temperature Reactor Technology Development Office/Advanced Gas Reactor Fuel Development and Qualification Program", PLN-3636, Rev 3, May 5, 2014
c. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2", SAND2012-4260 & SAND2012-4259, May 2012
d. ANL, "Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors", ANL-SMR-9, May 31, 2014
e. INL, "NRC Licensing Status Summary Report for NGNP", INL/EXT-13-28205, Rev 1, November 2014
f. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel", ANL-ARC-288, June 4, 2014
g. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014
h. INL, Personal communication with D. Petti, February 10, 2015
i. ANL, "Regulatory Technology Development Plan Sodium Fast Reactor, Mechanistic Source Term", ANL-ART-3, February 28, 2015
j. INL, Personal communication with H. Gougar, February 13, 2015
k. ANL, Personal communication with T. Sofu, March 20, 2015
l. ORNL, Personal communication with G. Flannigan, January 8, 2014

Description
Analytical Codes and Methods: Developing, optimizing, verifying and validating (V&V) the analytical methods and tools essential to performing a safety analysis is critical to a successful licensing outcome. Analytical techniques must be available to support system understanding and prediction modeling of important phenomena which may be quite unique to a particular design concept. Often, these phenomena are identified through expert panel elicitation. However, once a candidate parameter has been identified, objective data must be collected to support model validation. ART V&V programs should be coordinated with other efforts conducted across the entire regulated reactor community and with NRC staff to ensure interrelated research is performed in a comprehensive and complimentary manner that serves the largest possible universe of end users.

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 the calculational envelope required to analyze reactor systems**								
1.a	HTGR	Identify HTGR nuclear safety and performance envelope in terms of degrees of uncertainty regarding behavior and ability to predictively model.[a] Define the 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. Comprehensive and objective data must be provided that support a comprehensive analysis, along with associated uncertainties that accompany the characterizations.	Medium	Challenges still remain in abilities to model certain HTGR phenomena that influence significant safety parameters. Modeling these phenomena should be improved to quantify effects on core safety and performance parameters. The thermal fluid phenomena inadequately characterized include: air ingress after pipe break and blowdown; steam ingress after steam generator tube rupture; performance of passive vessel cooling system (air or water-based); heat transfer between blocks and across the core–reflector interface in pebble bed reactors (core heat transfer); extent of bypass flow between 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.[i]	Major scenarios required for the HTGR safety analysis have been identified, key phenomena and figures-of-merit have been documented, and a model validation matrix has been formulated. Related testing is underway at Oregon State University (OSU) High Temperature Test Facility (HTTF) to address air ingress, ANL Natural Convection Shutdown Heat Removal Test Facility (NSTF) to address vessel cooling performance, and at the HTTR facility (Japan) to provide physics data (rod worth, reactivity coefficients), pressure loss transient data, and performance of vessel cooling system data.[e] Model development, benchmarking, and uncertainty analysis of coupled neutronic/thermal fluid simulators will establish and characterize the uncertainty in baseline core modeling capability. In 2016, benchmarking projects will continue and some university (Nuclear Energy University Program [NEUP]) work will be performed in bypass flow, air ingress, and core heat transfer studies.[i]	Medium	Key R&D is underway. Licensing priority is established recognizing planned and underway work which includes: complete test plans for HTTF; vessel cooling studies in NSTF (both water and air-cooled); and plenum-to-plenum heat transfer studies.[e] Other priority work includes: 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.[i]

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 performance envelope in terms of the involved phenomena, degrees of associated uncertainty, and ability to predictively model. Define the scenarios that facilitate license reviews and identify key phenomena and figures-of-merit in importance scenarios of interest.[j]	High	Defining the plant safety performance envelope is an essential input to regulatory reviews. In addition to design information, this involves data concerning reactor core physics, primary and intermediate heat transport system thermal-fluids, safety metrics, physical processes during normal/off-normal/accident conditions, and the capabilities/limitations of analytical models. The information must be tailored to address specific design features and approaches affecting reactor safety.	High	Although many key parameters concerning the SFR design can be quantified from historic EBR-II and FFTF operational and safety testing experiences, some gaps in certain phenomena understanding may exist, particularly in relation to design features that depart from past experiences.[b]	Recovery of heritage EBR-II and FFTF operational performance and safety testing data is underway under DOE-NE's ART program. These efforts include recovery and retrievable archiving of data from EBR-II fuels irradiation experiments, EBR-II, FFTF and TREAT safety tests, and EBR-II and FFTF component reliability information.[h]	Medium	While extensive performance is available, the quality and completeness of the data must be confirmed against emerging designs. Medium licensing priority is assigned due to high regulatory importance, high state of existing knowledge, and a presumption that emerging designs will attempt to stay within existing data boundaries.
2	**Define evaluation model capable of performing required calculations encompassed by calculational envelope**								
2.a	HTGR	Identify/develop core and plant simulation tools of appropriate fidelity geared toward 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.[a]	High	Power reactors are licensed after demonstrating compliance with specified safety limits. Some limits are easily identified and predicted while others require complex models for accurate evaluation. Such modeling typically applies a complex mathematical representation of the system. Many different models can be combined into a common computer code to represent major system phenomena. Complex codes used for regulatory safety analysis must undergo a detailed assessment to demonstrate they are appropriate and reliable for the proposed application.[f]	Medium	HTGR fuel design and related specifications historically assumed design margin factors of 4 for fission gas release from the core and 10 for metallic fission product release from the core. These assumptions allowed fundamental design physics to be either ignored or approximated during analysis to degrees that makes it difficult (or impossible) to license a plant today. Predictive model development and validation is indicated to resolve this issue.[a] Neutronic phenomenon yet to be fully characterized in existing 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); extent of non-local fission energy deposition. However, inability to model these phenomena is not seen as a major barrier to licensing. Lack of knowledge of thermal fluid behavior is more significant.[i]	Adequacy of prior assumptions depends on outcome of the AGR fuel fission product transport data development tests, AGR fuel qualification tests, and AGR fuel fission product transport code validation tests. The ART Methods R&D program is geared towards using and refining existing software tools unless it is shown that capabilities of those tools are inadequate for design and licensing.[a] Data from AGR test program will significantly reduce uncertainties in modeling fission product transport through fuel compacts and blocks. Additional tool development which couples heat transport and 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 necessary experimentation and indicates source term sensitivity to various factors.[i]	High	HTGR fuel specification and analysis capabilities are key licensing concerns. Significant topical information is currently available and AGR test completion will extend the knowledge base. NOTE: ART methods R&D planned from 2016 to 2021 in support of VHTR is expected to fill many modeling capability gaps.[i]

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.b	HTGR	Develop new seismic analysis methods, such as an integrated seismic, structural, and systems-analysis approaches, to broaden applicability of existing seismic soil-structure interaction (SSI) computer codes to include deeply embedded or buried structures. Modify computer codes as necessary to address this need.[f]	High	Some advanced reactor designs (including HTGRs) call for reactor and steam generator systems to be built partially or completely below grade using deep embedments . Analysis of possible seismic events requires that SSI effects for deeply embedded structures be examined for resulting influence on an analytically predicted seismic response. An assessment capability for such regulatory analysis must be available.[f]	Low	Current soil-structure interaction 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 that are deeply or completely embedded below grade is needed.[f]	Seismic effects 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 incomplete.[c] R&D in this area is applicable to multiple reactor designs that employ a deep embedment. While initial work on the topic may be underway for LWR-derived SMR designs that employ embedment, no ART research tuned specifically for non-LWR designs is underway.	Medium	HTGR seismic analysis methodology has not been identified or reviewed by NRC for use. Seismic analysis specific to the HTGR will require some design development that is not yet planned for resolution. However, this delay is not yet seen as impacting a critical licensing timeline for the overall R&D plan.
2.c	SFR	Develop and validate an analysis code system as a regulatory acceptable primary and intermediate heat-transport system modeling and safety analysis tool.[a,g] Maintain code system by preparing V&V test matrixes and detailed documentation (not only for V&V outcomes, but for detailed code descriptions) to facilitate regulatory reviews. Improve modeling capabilities to include interfaces for high-fidelity multi-physics methods to reduce uncertainties in modeling of integrated neutronic, thermal, hydraulic, structural phenomena, and characterization of processes that could contribute to a mechanistic source term.[d]	High	Power reactors are licensed by showing compliance with specified safety limits that may require complicated modeling evaluations. Codes used for regulatory purposes undergo a detailed assessment to demonstrate appropriateness and reliability in the application.[f] Properly characterizing important elements such as complex thermal mixing and possible changes to boundary conditions that could disrupt system performance may require high-fidelity CFD tools. Modeling processes that are likely contributors to the SFR mechanistic source term are also essential to safety analysis.	Medium	Current SFR safety analysis codes are R&D tools that have yet to undergo regulatory review and receive NRC acceptance. While the roots of these tools date back decades, these codes have undergone continuous use and update as R&D tools. However, since they have not been used in a rigorous regulatory environment, all of these tools lack the quality assurance controls expected to support licensing decisions.[i]	The SAS4A/SASSYS-1 research code could serve as a central tool in multiple SFR licensing safety analysis scenarios. Usage in this capacity requires the code to be updated, verified, placed under configuration management, and reviewed by NRC staff for regulatory acceptance. Existing data, which support the code, need to be assessed in terms of its adequacy for a full spectrum validation. A modernization and maintenance program for SAS4A/SASSYS-1 is underway under DOE-NE's ART program.[d] Other codes that may exist, including those from foreign sources which may be considered for use as a potential safety analysis resource, also requires detailed qualification and review prior to regulatory use.[b]	High	Until uncertainties in key analysis code status (e.g., SAS4A/SASSYS-1) are defined in relationship to plant design safety conditions, the codes are reviewed by NRC staff for regulatory acceptance, and research is planned to address gaps pertaining to that acceptance, this issue should be considered a high priority regulatory concern.
2.d	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.[b]	High	Qualification of an SFR fuel design and performance analysis tools) such as LIFE-METAL) is essential to completing license reviews and enabling a broad understanding of mechanistic source term.	Medium	LIFE-METAL is an established R&D fuel performance code that is recognized as a likely candidate for regulatory use in SFR safety analysis. The underlying validation database and documentation related to the model needs to be updated.[b] NRC staff has not yet reviewed the code for the purpose 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 the DOE-NE's ART program.[b] Continued development and maintenance of LIFE-METAL code is not currently supported by DOE-NE.[h]	High	Validated fuel performance analysis codes are essential for licensing success. SFR code maturation currently hinges on recovery and qualification of heritage EBR-II and FFTF fuels irradiation experimental data. Until plans are established to develop a qualified fuel code, this topic is a major 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
2.e	SFR	Update the MELCOR code with a CONTAIN-LMR module to cover phenomena related to sodium pool and spray fires, and sodium-concrete interactions.[b,d]	High	NRC relies upon a suite of analysis codes that are used to support LWR licensing decisions. The severe accident analysis code is MELCOR. Integration of an SFR containment design analysis capabilities (CONTAIN-LMR) into MELCOR by adding to its radionuclide tracking capabilities will be important to support later regulatory MST evaluations.	High	MELCOR is a well-established regulatory LWR analysis code with a controlled configuration. Integration of the CONTAIN-LMR module, which is not currently supported in the U.S., into MELCOR would create a well-maintained and regulatory acceptable capability for radionuclide tracking, structure performance and containment response analyses.[b]	The sodium-fire and sodium-concrete interaction analysis capabilities of CONTAIN-LMR are currently being modified for integration into MELCOR under DOE-NE's ART program. However, MELCOR is an NRC code and adjustments require involvement of NRC staff and NRC code configuration control authorities.	Medium	Although updating MELCOR with CONTAIN-LMR capabilities is important, it is an adjunct to establishing other essential SFR specific safety codes (e.g., Items 2c and 2d discussed above) needed to support MST calculations.
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.[e] 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 a critical concern to a safety analysis. Data used to support the models must be of high quality (i.e., meeting applicable quality assurance standards), complete, and able to address safety margins. Data that support the model are subject to review and acceptance by NRC safety reviewers before associated analysis tools are employed in licensing-related decisions.	Medium	Scenarios required for the (NGNP) HTGR analysis have been identified. Development, verification, and validation of thermal, neutronic, and fluid codes cannot be completed without a parallel experimental program to supply these new tools with essential data that envelope anticipated HTGR 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.[e]	Research priorities emphasize establishing key test facilities for conducting integral experiments in the High Temperature Test Facility (OSU), refurbish and operate the Natural Convection Shutdown Test Facility (ANL) for investigation of ex-core heat removal, perform bypass and air ingress experiments with associated computational fluid dynamics 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 multiphysics HTGR analysis capabilities on the MOOSE platform is underway.[e]	Low	Necessary R&D to address this topic is underway. Natural Convection Shutdown Test Facility and High Temperature Test Facility tests are scheduled for execution between 2015 and 2021, depending on gaps in the validation matrix. Acquisition of other necessary data is underway and in many cases nearing completion. Although an essential licensing concern, this activity is prioritized lower in recognition of the advanced state of required research.
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 essential to successfully completing a safety review.	Medium	The SFR R&D design and analysis codes that exist today have not been reviewed and approved for use in regulatory process. Research data may or may not be of sufficient coverage and/or quality to validate their use in regulatory applications. State of knowledge is considered medium but is contingent on results of a detailed analysis related to legacy data gaps.[j]	Retrieval of operations and safety testing data from EBR-II, FFTF and TREAT is underway under the DOE-NE's ART program.	High	No major activity other than data retrieval-related work is underway. No work is yet planned regarding a code validation matrix for SFR technology. This lack of planned resolution makes the topic a high licensing concern.
4	**Verify adequacy of evaluation models using an approach conformant with NRC Regulatory Guide 1.203**								
4.a	HTGR	Perform calculations and necessary evaluations of model adequacy using NRC-accepted validation practices and procedures.[e]	High	The design and safety analysis tool qualification must 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	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 OECD MHTGR350 Benchmark of steady state, transient, and lattice codes for prismatic reactors and the IAEA Uncertainty Analysis Methodologies for High Temperature Reactors.[e]	Medium	This activity is of licensing interest and is evolving with respect to available knowledge. An approach exists to verify the adequacy of HTGR safety models.

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.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 determine the ability of the evaluation model or its component devices to predict outcomes according to appropriate experimental behavior.	Low	SFR safety analysis tools that are candidate for use in licensing are primarily R&D codes or were developed for use by regulatory agencies outside the US. All codes 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 of these codes, the important regulatory question centers on what will be required by the NRC to assure future regulatory acceptance of these tools.[j] This issue will require interactions with the NRC staff to address that question.	Various SFR reactor designs have used computer codes that are maintained by the DOE national laboratories, such as DIF3D, REBUS, MC2-2, VARI3D, ORIGEN-2, for neutronics, SAS4A/SASSYS-1 for safety analyses, NUBOW-3D for core restraint system design, SOFIRE for sodium fire analysis, and SWAMM for steam-generator tube rupture assessments. Although analysis capabilities of these codes have been established, simulation of neutronics, thermal, structural, fuel behavior, and hydraulic effects have yet to undergo a rigorous V&V and QA process.[f,g,j]	High	Once a comprehensive list of necessary analytical tool has been established, along with existing capabilities and associated gaps, existing (R&D) codes can be compared to RG 1.203, and plans established to address deficiencies. Until the list is established, the topic is a significant concern.

a. INL, "Next Generation Nuclear Plant Methods Technical Program Plan", Document ID: PLN-2498, Rev 3, December 21, 2010
b. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2", SAND2012-4260 & SAND2012-4259, May 2012
c. INL, "Graphite Technology Development Plan", PLN-2497, October 4, 2010
d. ANL, "Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors", ANL-SMR-9, May 31, 2014
e. INL, "NGNP Program 2013 Status and Path Forward", INL/EXT-14-31035, Rev 0, March 2014
f. NRC, "Advanced Reactor Research Plan", ML020730737, March 2002
g. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel", ANL-ARC-288, June 4, 2014
h. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014
i. INL, Personal communication with H. Gougar, February 13, 2015
j. ANL, Personal communication with T. Sofu, March 20, 2015

Description
Core Heat Removal Advanced reactor R&D must address issues of safety in core heat removal and expand those issues to include heat-related topics may that otherwise influence plant performance. The role of SSCs important to safety during AOOs, DBAs, and BDBEs, and how those elements relate to core heat removal, must be precisely understood and merged into a comprehensive safety design basis. Research in support of this analysis becomes more important the further an advanced reactor design departs from traditional LWR solutions. For instance, a liquid metal fast reactor operated at close to atmospheric pressure and at temperatures far below the boiling point of coolant metal will not lead to the same type of depressurization, coolant boiling, and loss of coolant accident (LOCA) that is experienced in LWRs in the event of coolant leakage or pipe break. This, in-turn, makes the LWR emergency core cooling system (i.e., a coolant injection capability under high and low pressure conditions) unnecessary in a liquid metal reactor. However, other support systems may be required to assure cooling capability is maintained.

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	Reactor Core Heat Removal								
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 adequately removed to the ultimate heat sink using air or water as the primary heat transfer medium. Assessment of system capabilities should consider influences resulting from atmospheric effects, system degradation factors, and system failure potential while in passive heat removal mode.[a]	High	Modular HTGR design efforts supported by ART presume a passive heat removal system is employed to ensure core heat is safely removed during licensing basis events. Demonstrating the effectiveness and reliability of this system to operate when required supports the overall safety basis for these simpler and more passive designs.[a,b]	Medium	The system responsible for this function in forthcoming modular HTGR designs is called the Reactor Cavity Cooling System (RCCS). Conceptual and preliminary designs using air as the cooling medium have been developed by several HTGR designers. A water-based RCCS concept has also been developed. Up to now, capabilities to physically test such systems have been limited. Additional data is necessary to firmly establish RCCS passive capabilities and provide data for analytic code V&V. However, because past HTGR projects like Fort St Vrain have been licensed with previous core cooling system information that is still available, this activity is established at a medium level of knowledge.	Testing of a scaled air-cooled RCCS (based on the General Atomics design) is underway at ANL's NSTF within the Thermal Reactors area of the ART program. Testing scope includes consideration of system degradation, weather effects, etc. When air-cooled verification test data collection is completed in 2015, the test facility will be available for reconfiguration to perform water-cooled RCCS testing; reconfiguration details are now being developed for a test plan that concludes in 2019.[e] (NOTE: A parallel study concerning characterization of potential passive system failure is underway within the ART Licensing Technical Area PRA activities group.)	High	Air-based RCCS testing at NSTF is near completion with data analysis scheduled for FY2016[c]. A water-based RCCS test protocol at NSTF remains to be implemented. Since a water-based RCCS system is a viable deployment option, completion of the water-based RCCS test plan at NSTF should be a high priority.
1.b	SFR	Develop capability to test, evaluate, and validate key design parameters and performance capabilities for a passive SFR core heat removal system. This system may utilize air or water 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 passive system.[d]	High	The SFR designs currently being developed are understood to rely on some type of passive heat removal system to ensure core heat remains at safe levels during licensing basis events. Systems such as these represent a key contributor to the overall safety basis for the SFR design.	Low	The SFR systems likely to be used to address core heat removal in a sodium pool-type design arrangement include multiple loops where each loop consists of a submerged in-vessel direct reactor auxiliary cooling system (DRACS) heat exchanger (e.g., twisted tube heat exchanger).[d] A reactor vessel auxiliary cooling system (RVACS) may be similarly utilized to remove heat from the reactor containment vessel using natural air convection.[b] Physical testing of such systems have been limited. Additional data reflective of current design trends are required to establish performance capabilities and allow V&V of the analytical codes used to assess system performance.	Very compact intermediate and DRACS heat exchangers are needed to reduce overall size of the primary reactor. R&D will be needed to bring the natural circulation performance of DRACS decay heat removal systems used in safety-related applications to a sufficient level of maturity to allow use in a sodium reactor environment.[d] Plans that support this R&D effort remain to be established and should look to the possible benefits provided by the RCCS testing being conducted at the ANL NSTF (see Item 1.a above).	High	Passive emergency decay heat removal capabilities will require natural convection cooling capabilities (to the atmosphere) as provided by multiple DRACS.[d] Until reliability and performance of such systems can be demonstrated for all design conditions, testing and validating these capabilities represents a key licensing topic. Data obtained from the NSTF tests will also be applicable to the validation of code simulating passive heat removal from the vessel.

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

a. INL, "Modular HTGR Safety Basis and Approach", INL/EXT-13-30872, January 2014

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

c. INL, "Baseline Concept Description of a Small Modular High Temperature Reactor", INL/EXT 14 31541, Rev 1, May 2014

d. ANL, "Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel", ANL-ARC-288, June 2014

e. ANL, Seminar – "Status of RCCS Alliance and Design Planning for Water-based NSTF, Argonne National Lab", February 24, 2015

Description
Material Analysis Licensing success in new reactor technology also depends on the outcome of materials science research. New or existing materials used in applications not been previously reviewed or approved by NRC may require R&D to establish a technical basis for regulatory approval. A sound technical basis will be necessary to evaluate, verify, and confirm the safe use of material including an understanding of failure modes. Time-dependent failure criteria for materials must be developed to ensure safety and adequate operational life of components. Development of applicable codes and standards may be sponsored by organizations such as the ASME BPV code for advanced reactors to establish a common understanding of structural materials in current design methodologies. The range of neutron flux, operating temperature, material compatibilities, and corrosive conditions that accompany some advanced reactor design concepts can challenge existing knowledge about the limits and effectiveness of metal and non-metal materials used in safety SSCs. The composition of these SSCs, their fabrication and the context of their application, and the resilience with which they can be predicted to withstand the rigors of use and intrinsic issues like creep and radiation effects, is a typical licensing concern which can only be addressed through appropriate R&D. It is important to remember that materials research in support of a specific reactor technology should be planned and performed wherever possible to provide insights applicable to other types of reactor design.

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	Develop supply sources and qualify nuclear grade graphite for use in HTGR core. Generate test data from irradiated sample examinations to more precisely predict material properties and behaviors in support of safety analysis code development.[c]	High	Nuclear grade graphite is a neutron moderator and key structural element in HTGR fuel blocks and is relied upon to preserve core cooling configuration. Understanding and predicting graphite behavior and responses during normal operations and accident conditions is essential to successfully completing the NRC safety review process. Since "historic" nuclear grade graphite sources no longer exist, new sources of supply must be qualified for use in HTGR applications.	High	Extensive information already exists on past sources of nuclear-grade graphite. Irradiation induced creep is currently the primary 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. Test data are also necessary to create an enhanced multiscale graphite modeling capability.[c]	Historic information on nuclear grade graphite and qualification for use in HTGRs is available. Recent R&D includes establishing ASME BPV Section III, Division 5 code rules for nuclear grade graphite which will require update as new graphite data becomes available.[f] Supplemental graphite irradiation experiments and characterizations are now underway as part of the Advanced Graphite Creep (AGC) program; these tests are detailed in a NGNP graphite technology development plan.[c]	Medium	Although extensive information is currently available, additional qualification testing of nuclear grade graphite is warranted to further material behavior predictive capabilities. R&D to qualify new graphite materials and refine analysis tools is currently underway in connection with AGR test plan and is essential for licensing success.
1.b	HTGR	The reactor pressure vessel (RPV) must be designed and fabricated to ensure vessel safety function is adequately maintained during anticipated design conditions. Generate confirmatory data that support related design conclusions about RPV safety.	High	The HTGR RPV is a safety-grade system relied upon to provide structural integrity and preserve the cooling geometry of the core. Regulatory standards require it to be constructed of durable materials and compatible with other reactor materials and expected plant conditions. In HTGRs, a key RPV concern involves construction material response to high temperatures.	High	The ART technology development plan details the R&D required to design and license a HTGR RPV (assuming SA-508/SA-533 is the material of construction).[e] Sufficient data is available to validate mechanical properties of SA-508/SA-533 steel but additional data are needed relative to long-term aging behavior at HTGR vessel temperatures and to understand environmental effect differences from LWR experience. Data are also needed on the effect of impure helium on long-term corrosion and mechanical properties.[e]	As a result of current design approach understandings, ART R&D and ASME BPV Section III code development efforts have 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 be subject to design consideration but will likely not be used in the initial technology demonstration plant. While knowledge of SA-508/SA-533 is good, if the design changes to an alternative material, substantial R&D effort may be required to develop the qualified data and code information necessary for that material.	Low	Guidance contained in the ASME BPV Sec III code supporting HTGR design and construction has been developed but remains to be endorsed by NRC. NOTE: Unless the applicant requires RPV construction with materials that withstand higher temps than now presumed, additional near-term research on this issue is not considered an outstanding concern.

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.c	SFR	Develop radiation response data for key metallic construction materials to adequately support their use in SFR cladding and ducts. This is to be done in close relationship with SFR fuels development. Perform research to supplement knowledge gaps in existing database. Ensure data addresses all plant design conditions.[a]	High	Factors that may significantly affect fuel performance, integrity, and the mechanistic source terms during normal conditions and design basis accident events are evaluated during the regulatory safety analysis. Objective test data and an ability to predict long-term material performance are necessary to support the evaluation. A thorough knowledge of material properties is used in applications like fuel cladding and ducting, along with future performance and quality issues associated with component fabrication is necessary.	Medium	There are currently two alloy classes with enough radiation response data to consider in SFR 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 the high-radiation conditions of SFRs. Current SFR fuel cladding and duct material knowledge (and fabrication experience) is part of the legacy SFR information bounded by the EBR-II and FFTF operating envelope. These data boundaries are probably insufficient for efficient power generation plant design.[a] 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 designers remains 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, the limited amounts of published data concerning this class of material precludes declarations that the material is suitable for use in a SFR.[a] Necessary research is expected to confirm quality applications of new fuel cladding materials for all prospective SFR vendor designs.[b]	Medium	All SFR design vendors will need to qualify new cladding and duct materials.[b] Additional R&D will be needed but that planning is contingent on vendor design choices and the scope and quality of existing legacy data. Until gaps are quantified and R&D plans developed to address gaps (which could require fast neutron irradiation experiments), this activity should be considered a potentially significant issue.
2	**Advanced Reactor Material Applications**								
2.a	HTGR	Ensure data and information is available to define and predict performance of materials used to assist transport of reactor heat to external heat sink. Include systems related to steam generator, intermediate heat exchanger, the core, and other related SSCs.	High	NRC safety analysts' must thoroughly understand the means by which thermal energy generated by the reactor core is safely transported to the external heat sink. Only the external heat sink is credited for plant safety during a safety review. All factors that may affect core heat transfer capabilities during normal operational and design accident conditions are to be characterized.	Medium	Research objectives specifically related to the high-temperature applications of the HTGR steam generator, intermediate heat exchanger (IHX), the core barrel, and core internals such as control rod sleeves, are addressed in a technology development plan.[f] The plan was established to add material performance data and develop models which were inadequate for many of the high temperature alloys that may be required by HTGR codes and standards. Research activity is currently limited pending applicant design decisions.	Improved understandings are necessary concerning environmental effects and thermal aging of high-temperature alloys. Welding and joining procedures and certification of various components are still needed that address very thick plates and thin sheets. Inspection parameters must be defined and developed. Heat exchange system details and performance requirements cannot be finalized until applicant specifies required heat load performance envelope.	Medium	There are currently no materials recognized as available for use above 800°C as the allowable life of high temperature materials is not sufficient to support desired design life.[g] Applicant design decisions are necessary to support issue resolution.

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 which 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.[a]	High	Using metallic metals like sodium for reactor cooling creates significant industrial hazards concerning potential chemical incompatibility, reactivity and fire. Historically, NRC has consistently sought to minimize these 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 a NRC determination that plant safety will not be unacceptably compromised when using sodium technology.	Low	An initial expert elicitation of phenomena relevant to sodium technology safety, the criteria important to a safety evaluation, and the status of phenomena knowledge has been done.[a] Twenty-six gaps of varying degrees of importance were identified in establishing a safety case. While existing material performance knowledge may be adequate for important design issues that remain within the historic SFR plant operating envelope, the quality and comprehensiveness of that data remains to be confirmed against emerging designs. Significant research activity will be necessary to develop data that supports predictive effects modeling and also concerning plant conditions that exceed the legacy data envelope boundary, including accidents.	Facilities exist (or can be revived) to support laboratory scale sodium technology tests (including pool fires).[a] New methods and instrumentation must be developed to perform necessary inspections and core tests in the opaque and corrosive sodium environment [also see I&C entries in Table 5]. Predictive analysis tools must also be developed. Current sodium technology knowledge is primarily constrained to legacy SFR data from prior plant operations and the information generated from foreign sources. While further R&D of sodium technology is not believed a long lead-time item as might be associated with irradiation testing, the regulatory implications and demonstrations required for metallic sodium use will require extensive prelicensing interaction with NRC and may guide research needs in directions that are currently unforeseen. Research approaches should be carefully planned and performed in conjunction with inputs from NRC staff.	High	The presence of metallic sodium represents a major plant industrial vulnerability with potential implications in nuclear safety. Extensive prelicensing interaction with NRC is strongly advised to ensure requisite R&D is planned to adequately address safety review concerns. A detailed review of the regulatory framework regarding sodium technology is also recommended. A technology development plan specific to sodium is advised to guide licensing-related research needs.
3					**Development of Codes and Standards**				
3.a	HTGR	Although a license to build and operate a nuclear reactor is granted by NRC, construction of key structural components is expected to comply with Section III of the ASME BPV code. Ensure the ASME code (or an acceptable equivalent) is developed and updated to adequately represent HTGR construction issues, including the RPV.	Medium	Developing and adhering to recognized national standards and consensus codes as part of licensing is important to facilitate a safety review. However, NRC is reluctant to endorse new industry codes and standards for reactors in a "piecemeal" fashion and generally waits until an application is submitted which cites the new code or standard before evaluating and endorsing it for use. Thus, the first-of-a-kind reactor technology applicant bears added burdens in first assuring codes and standards are appropriately developed for the design and then justifying the code before NRC during application review.	High	Recently, HTGR rules have been initially developed to represent the current NGNP design and are contained in the new Division 5 within Section III of ASME BPV code.[d,e] However, while this code is relatively current, it is not yet active for use by applicants and has yet to be reviewed by and endorsed by NRC staff. Prospective applicants must review and confirm the proposed standard adequately represents their HTGR design.	When HTGR plant design work resumes, the plant designer will need to evaluate what additional code support will be needed from national organizations and re-establish appropriate levels of stakeholder engagement. 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 but formal approval of the codes need a vehicle (i.e., an application) to initiate refinements and a formal NRC review. Additional code work is probably needed but is contingent upon license application development. Given current status and need for applicant involvement, this issue is currently considered a low 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
3.b	SFR	Although a license to build and operate a reactor is granted by NRC, construction of key nuclear plant structural components in the U.S. also complies with Section III of the ASME BPV code. Ensure this code or a code equivalent is developed and updated to representatively address the construction of liquid metal reactor (LMR) technology and, more specifically, SFR technology.	Medium	Adherence to approved national standards and consensus codes as part of reactor design and construction is important to facilitate NRC reviews of a license application. The NRC has been reluctant to endorse new reactor codes and standards in a "piecemeal" fashion and waits until an application is before them citing the new code or standard. The first-of-a-kind reactor technology applicant bears a major burden in assuring these codes and standards are adequately developed and subsequently endorsed by NRC.	Medium	LMRs are addressed in the new Division 5 formed within Section III of 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 (perhaps significantly) vary with design approaches now emerging from prospective SFR vendors.[g] The code is not yet agreed to by NRC for use by an applicant.	The high temperature materials and design methods currently contained in the code were developed for SFR license applications but have not been significantly updated since the 1990's. Modern design methods need to be developed, demonstrated, and incorporated into the code. New materials with enhanced creep strength and life are needed to facilitate design.[g]	Low	ASME BPV codes for SFRs have been proposed but must be confirmed as representative of emerging design ideas and modified as indicated. Also, further SFR design maturity is needed to understand material selection and its application for qualification needs. Given current status and need for applicant involvement, this issue is considered a low licensing priority.

a. SNL, "Sodium Fast Reactor Safety and Licensing Research Plan, Vols 1 & 2", SAND2012-4260 & SAND2012-4259, May 2012
b. ANL, Personal communication with T. Sofu & C. Grandy, December 15, 2014
c. INL, "Graphite Technology Development Plan", PLN-2497, Rev 1, October 4, 2010
d. INL, "NGNP High Temperature Materials White Paper", INL/EXT-09-17187, Rev 1, August 2012
e. INL, "NGNP Reactor Pressure Vessel Materials R&D Plan", PLN-2803, Rev 1, June 14, 2010
f. INL, "NGNP Steam Generator and Intermediate Heat Exchanger Materials R&D Plan", PLN-2804, Rev 1, September 23, 2010
g. INL, Personal communication with R. Wright, February 27, 2015

Description
Instrumentation and Control Coming generations of nuclear reactors will take advantage of integrated digital control rooms and utilize new technological advances to support appropriately-sized and potentially smaller control room staffs. Plant designs are expected to push to much higher levels of automation. Multiple interconnected modular plants may require more sophisticated supervisory and control functions in both primary and support I&C systems. Support systems could include a physical interfaces and response capability with nearby industrial energy users. The I&C systems that are deployed must ensure they can directly measure, diagnose, and respond to the safety and operating parameters relative to conditions deemed important in the safety analysis. Research on digital I&C advancements will be needed to ensure new design are analyzed and confirmed of adequate reliability. This configuration involves new I&C requirements and equipment such as sensors, control units, and updated techniques in data integration.

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 plausible core irregularities such as local core hot spots, fuel misloadings, pebble flow anomalies, block-stack motions, and other related conditions.[a]	Medium	Previous NRC review of NGNP prelicensing 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 concern represents a preliminary staff opinion and it can be assumed that the initial HTGR applicant will be required to resolve this issue to the satisfaction of NRC.	Low	No "in-core" monitoring capability has yet been demonstrated as capable of performing reliably over extended periods under nominal HTGR core conditions. Reliance on "near-core" detection would likely create additional uncertainties regarding actual in-core TRISO fuel conditions and could potentially result in overly restrictive core operating limits to conservatively satisfy related functional radionuclide containment assumptions.	There is no integrated research effort underway or planned to address this concern. ART currently does have an I&C activity underway (Work Package AT-15OR230103) to develop the Johnson Noise Thermometry Monitoring concept; this effort may have limited applications to HTGR in-core condition monitoring. The DOE Nuclear Energy Enabling Technologies (NEET) program is working on high temperature sensors that may have applications but it is unclear whether that effort can address HTGR conditions.[b] High-sensitivity, high-temperature "micro-pocket" fission chambers and gamma thermometers have been considered a potential options for local power measurements, but only limited work has been done in the area so far.	Medium	The inability to accurately measure in-core parameters will create additional regulatory uncertainty regarding the results of predictive analytical models. This will likely result in the need to implement overly conservative plant operating limits to satisfy licensing requirements related to core performance, which may not be technically or commercially viable.
1.b	HTGR	Develop capability to reliably measure, monitor and control operation of HTGR RCCS for passive heat removal. These systems typically exhibit low flow and low pressure conditions during both normal and accident plant conditions	Medium	The RCCS is relied upon to maintain plant safety in modular HTGR designs during accident conditions. This system typically operates during all modes of plant operation. A capability to predict and monitor system availability and performance is a key issue for in the plant safety review. NRC has also identified this issue as a topic regulatory concern during its review of the GE-PRISM SFR design.[d]	Low	The current state of topical knowledge is largely limited to historic information and capabilities associated with previous HTGR design activities conducted by the industry. Additional RCCS technology demonstrations are nearing completion at the ANL's NSTF.	Insights into measurement and monitoring capabilities for these 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 made by the applicant.	Medium	The RCCS is the safety-related system relied upon for core heat removal during all design basis accidents. The ability to predict and monitor the system's performance during design basis events is expected to be a critical licensing requirement for modular HTGRs.

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 that can reliably measure key safety-related parameters such as flow and pressure in a liquid metallic sodium environment.[b,c]	Medium	Demonstrated capabilities in monitoring and measuring safety-related parameters in challenging environments like a liquid metal pool is critical to the safety review process. During reviews of the GE PRISM design, NRC staff emphasized the importance of "*...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.*"[d]	Low	The current state of knowledge is limited in this area and largely a function of historic SFR plant operations. Existing knowledge base is considered insufficient to address stated NRC concerns on the topic.	There is no integrated effort underway or currently planned to address research needs on this topic.[b] DOE's NEET program work on high temperature sensors can be cited but also indicates a need for a specific focus on the challenge in SFR applications. High-sensitivity, high-temperature ("micro-pocket") fission chambers and gamma thermometers have been previously considered as potential options for local power measurements, but limited work has been done in these areas to date.[e]	Medium	The inability to accurately measure key parameters in harsh environments creates additional regulatory uncertainty regarding the results of predictive analytical models, and the capability to monitor plant conditions during both normal operations and during design basis events.
1.d	SFR	Establish capability to reliably measure and monitor operation of RVACS-type passive heat removal systems in the SFR design. These systems typically exhibit low flow and low pressure conditions during both normal and accident plant conditions	Medium	NRC has indicated the significant of accurately monitoring the SFR's passive safety cooling system. "*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.*"[d]	Low	The current state of knowledge in this area is based on overall existing capabilities within industry, and demonstrations completed at ANL's NSTF.	Insights into measurement and monitoring capabilities for these types of passive cooling systems is being developed as a part of the RCCS testing currently underway in the NSTF at ANL. Additional testing and analysis may be needed to address RVACS-specific issues not adequately addressed by the RCCS testing.	Medium	The RVACS is a vital safety-related system relied upon for core heat removal during all design basis accidents. As noted by the NRC in a previous review,[d] the ability to predict and monitor this vital systems' performance during design basis events is expected to be an important licensing requirement for SFRs. The NSTF facility (on which the analogous HTGR RCCS experiments are being performed) will provide useful data for validating RVACS simulations.
2	**Advanced Surveillance and Diagnostics**								
2.a	HTGR	Establish reliable and accurate capabilities for measuring circulating radionuclide activity and the presence of moisture within the HTGR primary helium loop.	Medium	These parameters are key factors that contribute to the modular HTGR mechanistic source terms. They are also closely associated with HTGR-specific Specified Acceptable Radiological Release Limit (SARRDL) concerning TRISO-coated fuel which must be defined and maintained during normal plant operation.	High	Measurements of circulating radionuclide activity and moisture were required in connection with past HTGR operations at the Fort St. Vain plant. It has not yet been confirmed whether heritage methods and capabilities of measurement are adequate to confirm these factors and support the (yet-to-be-established) SARRDL requirements.	There is no ART-related research currently planned or underway in this specific area.	Low	Primary helium loop parameters are a primary constituent of the modular HTGR's mechanistic source term, so are a key licensing issue. However, these parameters have been successfully measured in previous designs, so this topic is not considered to be a significant ART program challenge at this time.

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.b	HTGR	Develop capabilities to monitor the integrity of reactor internals for modular HTGRs.	Medium	Assuring reactor internal integrity is critical in maintaining core configuration and the geometries required to assure passive cooling capabilities in the modular HTGR. Maintaining proper core configuration is essential to maintain safety under design conditions.	Medium	Internals integrity requirements and methods for confirmation are currently being assessed as a part of the AGC graphite qualification program (conducted at INL).[f] This work is being done in support of ASME code qualification. Results of this work may 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.	Low	Additional reactor internals integrity confirmation techniques may be required to be developed pending the results of ASME Code qualification efforts.
2.c	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 feedbacks relied upon to prevent core damage in unprotected accidents behave as expected.[b,c]	Medium	Reliance on passive feedback is a key safety characteristic of the SFR safety design and is relied upon in certain accident sequences. A capability is necessary confirm this passive safety feature is maintained as plant conditions may change.	Medium	The current state of topical knowledge in this area is limited to prior SFR technology R&D plant operations. State of development by SFR technology vendors is unknown.	There is no integrated effort underway or planned to address this research activity overall. An underlying capability is currently being developed in the Small Modular Reactor and Light Water Reactor Sustainability programs, but a SFR focus on work like this will be necessary to appropriately account for fast reactor specific phenomena, such as core expansion.	Medium	The development of the diagnostic capability to assess this core damage prevention measure must be completed to support the SFR's safety basis and licensing.
2.d	SFR	Develop methods and capabilities in detecting sodium leakage.[b]	Medium	Liquid metal sodium coolant properties add a dimension of chemical reactivity and material compatibility issues that must be considered when a sodium leak occurs. Understanding and controlling the potential for adverse consequences from sodium leakage must be fully considered in the evaluation of SFR reactor safety.	Medium	There is a substantial state of functional knowledge in this area based on historic experience with sodium handling and management techniques.	There is no ART research currently underway in this specific area.	Low	This important activity has been rated as a low near-term licensing priority due to the current state of available knowledge in this area.
2.e	SFR	Develop reactor internals integrity monitoring capabilities for SFRs.[b]	Medium	Reactor internals integrity must be routinely evaluated and confirmed to satisfy regulatory requirements. Significant technical challenges exist to routinely perform such monitoring in a sodium pool environment.	Low	The current state of knowledge is quite limited in this area, since most existing methods are based on LWR environments.	I&C technical development activities, such as under sodium viewing, are currently underway to begin to establish this set of capabilities in the opaque sodium environment. These activities currently included in the set of work package activities currently underway within the ART I&C area.[e]	Medium	Sodium pool environments are significantly different from the current experience base with water and must be addressed to establish viable methods for internals integrity monitoring and management.

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
3	Human-Machine Interface								
3.a	HTGR	The topics of the Human-Machine Interface (HMI) and Control Room Staffing for modular reactors are both generic industry issues that have been consistently identified as regulatory challenges for both iPWRs and advanced reactor technologies, including modular HTGRs. A plan for addressing these generic issues is not included in the RTDP at this time, pending progress through iPWR licensing interactions. The RTDP may be updated to include these items in the future.							NOTE: This item is included as a “placeholder” and will be regularly re-evaluated as the issue is addressed by the iPWR community. ART program activities initiated during re-evaluations will be added to this table and evaluated accordingly.
3.b	SFR	The topics of the Human-Machine Interface (HMI) and Control Room Staffing for modular reactors are both generic industry issues that have been consistently identified as regulatory challenges for both iPWRs and advanced reactor technologies, including SFRs. A plan for addressing these generic issues is not included in the RTDP at this time, pending progress through iPWR licensing interactions. The RTDP may be updated to include these items in the future.							NOTE: This item is included as a “placeholder” and will be regularly re-evaluated as the issue is addressed by the iPWR community. ART program activities identified during re-evaluations will be added to this table and evaluated accordingly.

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

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

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

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

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

f. INL, “Graphite Technology Development Plan”, PLN-2497, Rev. 1, October 4, 2010

Description
Safeguards and Security: A strategy for site security, including a “security by design” effort, should be developed early in the plant design process. Once that is done, a more detailed program can be developed to guide subsequent design decisions and determine whether any security and safeguards issues need additional developmental attention. These issues might include new sensor system, new approaches to fissile material inventory, and innovative methods in security response. A preliminary design security assessment is essential to ensure the integration of meaningful security and safety considerations into the design approach. This assessment also requires demonstrations that proposed approaches can be tested, challenged, and confirmed adequate.

Table 6. ART Research Regarding Safeguards and Security.

RESERVED FOR FUTURE USE

Description
Accident Sequences and Initiators: The scope of this area includes complex dynamic systems such as thermal-fluids, heat transfer, structural and neutronics modeling capabilities. It also considers the validation basis for simulations. Of particular interest are evaluations of accident analysis tools that are optimized to assess reactor safety. In addition, this area addresses the potential for modern modeling and simulation techniques to improve nuclear safety analysis using higher-fidelity, integrated multi-process tools. Based on the range of scenarios and phenomenology identified from the safety evaluation area, the codes and models area addresses the analytical capabilities and data required to adequately assess the safety implications of scenarios and phenomena. Accidents and associated phenomena important to establishing a safety case may be insufficiently known during early phases of technology R&D or described in ways that are not easily translated to risk-informed, performance-based considerations. A basic level of design understanding and systems analysis is required to support research. Planned research in this topic should strongly consider a technology neutral perspective wherever possible to avoid limitations such as those now exhibited by the LWR-centric regulatory framework.

Table 7. ART Research Regarding Accident Sequences and Initiators.
RESERVED FOR FUTURE USE

Description
Probabilistic Risk Assessment: Probabilistic risk assessment (PRA) is integral to future advanced reactor design certifications and license applications. The PRA provides a means to identified risk-informed safety insights based on systematic evaluations of risks associated with the design such that those insight can identify and describe the following: (a) the design's robustness, level of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events, and (b) the risk significance of potential human errors associated with the design. License applicants will need to have adequate design information available to objectively perform risk assessments to the satisfaction of safety reviews. As was noted by the NRC in their 2012 'Advanced reactor Policy Statement", advanced reactor designs are to be risk-informed, thereby making the PRA an important component in the overall design process. Limitations associated with advanced reactor PRA experiences can be anticipated with applications having new underlying hypotheses (e.g., treatment of passive systems), 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), inadequate failure data histories, and (perhaps most importantly) to the design, materials, systems, and safety approach. The NRC staff will determine if the technical adequacy of the PRA is sufficient to justify the specified results and risk conclusion contained in the license application.

Table 8. ART Research Regarding Probabilistic Risk Assessment.

Table 5 - ART Research Regarding Probablistic 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		**PRA Framework for Advanced Reactors**							
1.a	HTGR and SFR	A key area of the PRA strategy for advanced reactors is the development of methodologies and tools that will be used to predict the safety, security, safeguards, performance, and deployment viability of advanced reactor technologies. This activity is developing quantitative methods and tools, and the associated analysis framework, for assessing a variety of risks. These risks will be focused on advanced reactor designs and operational strategies as they relate to the technical basis behind the characterization of the safety case supporting licensing.	Medium	The development of safety models for advanced reactor margin determination will provide a safety case that describes potential accidents and design options (including postulated controls), and supports licensing activities by providing a technical basis for the safety envelope.	Medium	ART Program work on PRA Framework development has been underway since FY2013. That work focuses on model development, identification of phenomena, and evaluation of demonstration problems to establish methods for integrating risk results and insights. This work also examines 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.[a,b]	Development and implementation of safety assessment methods requires new analytic methods or adaptation of traditional methods to the unique design and operational features of advanced reactors. The initial demonstration phase of those framework attributes is planned for completion in FY2016. These framework tools will then be available for assessment by external stakeholders as they work to develop their specific design and licensing strategies.	Low	The use of advanced PRA framework methods early in the design process is expected to provide reactor developers with key design insights and better characterization of safety margin. This is considered a low licensing priority at this point until design vendors determine a path for proceeding with the next phases of framework development and implementation in their respective licensing plans.

a. INL, Summary of "Advanced Small Modular Reactor (SMR) Probabilistic Risk Assessment (PRA) Technical Exchange Meeting", INL/EXT-13-30170, September 2013

b. INL, "A Framework to Expand and Advance Probabilistic Risk Assessment to Support Small Modular Reactors", INL/EXT-12-27345, September 2012

Description
Structural Analysis: Structural analysis tools for LWRs are mature, standardized, and benefit from an extensive application database. It is unclear whether these tools can be used with minimal or no modifications for confirmatory structural analysis in non-LWR applications. Design variations between technologies may necessitate a tailoring of structural analyses capabilities. Additional capabilities will probably be necessary regarding the qualification of seismic isolators. Some advanced designs, such as the modular HTGR, presume the use of deep embedments that envelop the reactor core and associated heat exchange systems. Since existing seismic tools presume the plant intersects closer to the earth at grade, new seismic analysis tools will be necessary to support not only assessment of seismic impact to below-grade safety SSCs but also aid in developing the seismic isolation systems necessary to assure appropriate plant response during a seismic event. Such tools have not been used in a NRC reactor licensing action. This is an example of a technology R&D opportunity that can potentially crosscut multiple reactor technologies including small LWR designs that seek to use deep embedments.

Table 9. ART Research Recommendations Regarding Structural Analysis.
RESERVED FOR FUTURE USE

Description
Human Factors: Advanced reactors may present new operational and maintenance challenges substantially different than current practice. Changes in future approaches may be extensive and include modernization of large-scale and remote control rooms and use of computer-based technology as part of a digital I&C upgrade program. These changes can result in substantial modifications to alarms, controls, and displays associated with safety SSCs. Additional consideration must be directed towards making a functional requirements analysis and function allocation when considering human factors in design. Functional requirements analysis is the identification and analysis of those functions that must be performed to satisfy plant safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the public. 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). Procedures are expected to become more computer-based and may even control safety response actions in an automated manner with the operator only monitoring functions, bypassing the automation only when conditions dictate. Different staff training and qualification programs will be needed to maintain digital systems and focus decision-making around monitoring and bypassing automatic systems rather than directly controlling them through active operator intervention.

Table 10. ART Research Regarding Human Factors.
RESERVED FOR FUTURE USE

4. LICENSING PRIORITIES AND RECOMMENDATIONS IN ART RESEARCH

Successful technology development of modular HTGR and SFR technology is strongly linked with the ART program research activities identified in Section 3 tables. Successful nuclear plant design licensing, however, more heavily emphasizes attention to issues related to the questions of:

1. How does the fuel perform during normal and accident conditions, and what radionuclides are potentially released?
2. How do radionuclides released from the fuel make their way into the environment?
3. How is heat removed from the fuel to keep it inside an acceptable performance and radionuclide release envelope?

A review of individual research activities listed in Section 3 tables and the licensing priorities assigned them has produced eight recommendations for near-term consideration within the ART program. Implementing these recommendations and continuing to maintain a focus on long-term licensing success will enhance overall prospects of future success in commercial advanced reactor deployment.

The following recommendations are not presented in a suggested rank-order of priority.

4.1 Fuel Performance

RECOMMENDATION 1: Implement the following actions regarding recovered historic sodium fast reactor information:

1A. Continue recovery of legacy FFTF and EBR-II data

1B. Establish a data configuration control process for recovered information that can be used to support a safety case

1C. Perform a technical assessment of recovered legacy data to identify gaps in the performance envelope; start this analysis with metallic fuels-related information

1D. Develop a systematic approach to evaluate and qualify legacy research information for use in establishing a SFR safety case; verify that key legacy data can satisfy applicable regulatory quality assurance standards

Experimental SFRs have been constructed and operated in the U.S. Perhaps the most relevant sources of plant heritage information comes from experiences at EBR-II (located at the INL and operated from 1964-1994 using metallic core fuel), and the FFTF (located at the Hanford Site in Washington and operated from 1980-1993 using mixed-oxide core fuel). Both facilities were built by DOE and its predecessor, the Atomic Energy Commission (AEC), to demonstrate the viability of a sodium-metal cooled fast reactor. However, the 62.5MW(t) EBR-II design appears to share the greatest similarity with SFR concepts now being proposed for commercial deployment.[7] Neither of these reactors underwent a formal NRC safety review that will be required of a new commercial power plant.

Efforts are underway to recover and preserve EBR-II and FFTF legacy data under DOE-NE's ART fast reactor R&D program. To date, these efforts generally emphasized preserving information from data acquisition systems and hardcopy reports and entering it into modern electronic formats suitable for data later retrieval and examination.[21]

Recommendation 1A – Continue recovery of legacy FFTF and EBR-II data

Concurrent with recovery efforts, ensure documentation that describes test protocols and validation approaches used during tests are preserved and traceable back to affected data.

Recommendation 1B – Establish a data configuration control process for recovered information that can be used to support the safety case

Data and information important to safety should be managed according to applicable regulatory requirements as is placed in a new electronic format. Data integrity control is essential to the independent safety review process as without it key support information may be rejected by NRC staff, possibly resulting in a need for supplemental and confirmatory testing.

Recommendation 1C – Perform a technical assessment of recovered legacy data to identify gaps in the performance envelope; start this analysis with metallic fuels-related information

Identifying and addressing key gaps in the fuel qualification effort is a critical licensing concern. Accurately identifying these gaps will require interactions with the SFR vendor community to better understand the emerging design safety case. Interaction with NRC staff is also likely so as to clarify regulatory requirements and ensure available data will address regulatory criteria. Because the SFR metal fuel variant is most likely (of available options) to undergo an initial NRC safety review, it is recommended the assessment start with this fuel type and stress development of further understanding concerning fuel performance under accident conditions. [21]

Recommendation 1D – Develop a systematic approach to evaluate and qualify legacy research information for use in establishing a SFR safety case; verify that key legacy data can satisfy applicable regulatory quality assurance standards

Test data and operational information generated by past technology development projects can be assumed to have been generated using good scientific principles and research practices in effect at that time. Quality assurance measures used of those activities remain to be confirmed as adequate and appropriate to the standards expected for data used in a contemporary independent safety review. Should information important to the technology safety case be found deficient in a key quality attribute, efforts to "upgrade" data might require confirmatory testing. Since these data must be accepted by NRC prior to use in a safety review, interaction with both technology vendors and NRC will likely be required.

RECOMMENDATION 2: Initiate preliminary planning for additional SFR fuel tests to address gaps identified under Recommendation 1C

At this time, the extent of supplemental SFR fuel test information that must still be acquired is not well established. Experience with U-Pu-Zr in metal fast reactor fuel is known to be limited should that be determined as relevant but binary (U-Zr) metal alloy fuel knowledge is much more extensive. Relatively little may be known about fuel performance margins outside of the nominal operational envelope of the data sets generated by EBR-II and FFTF.[21] Since commercial reactor designs are required to demonstrate fuel performance outside the boundaries of normal operation, additional testing may be necessary.

It is worth noting that fast neutron irradiation test facilities are not available domestically and are limited at foreign locations. It is recommended that a research plan to develop and justify the fuel qualification portion of the SFR safety case concerning the initial plant be further developed and presented to NRC staff as early as possible for their review and feedback.

RECOMMENDATION 3: Complete the Advanced Gas Reactor Test Plan as described in PLN-3636 and the Graphite Technology Development Plan as described in PLN-2497

The ART Very High-Temperature Reactor (VHTR) Technology Development Office (TDO) operates the AGR fuel test program. This program was established to expand the modular HTGR fuel test 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. Further details concerning scope can be found in PLN-3636, "Technical Program Plan for the Very High Temperature Reactor Technology Development Office/Advanced Gas Reactor Fuel Development and Qualification Program".[9]

Data to be gathered at AGR test plan completion are essential to fuel performance predictions and MST calculations. Indeed, the safety approaches developed for a prismatic-block HTGR core based on expected AGR test results have already been reviewed by NRC staff and found reasonable.[16]

Another closely related effort involves completion of the PLN-2497, "Graphite Technology Development Plan".[13] Graphite physically contains the modular HTGR fuel and comprises the majority of the core volume, thereby substantially influencing safety and MST calculations. While the general characteristics of nuclear grade graphite are understood, historic "nuclear" grades no longer exist; new grades must be fabricated, characterized, and irradiated to demonstrate that current grades exhibit acceptable non-irradiated and irradiated properties upon which the thermomechanical design of structural HTGR graphite is based. Data generated by completing PLN-2497 (sometimes referred to as the Advanced Graphite Creep [AGC] Program) are essential to completing a modular HTGR safety assessment.

The essential nature of these test programs with respect to licensing success is discussed in Chapter 3. However, please note that the "medium" licensing priority that may be assigned to certain AGR and AGC tests acknowledge the state of advanced research performance and expected progress in the test programs. Delaying or cancelling tests associated with the AGR and AGC programs would cause many of the licensing-related elements that appear in Chapter 3 tables to be re-prioritized as a high level of concern.

4.2 Radionuclide Transport Methods

RECOMMENDATION 4: Complete systematic development of modular high temperature gas-cooled reactor-compatible safety analysis methods and codes

Historic HTGR safety analysis codes relied on approximations and assumptions about physics behavior of fuels, materials, and coolants. These factors are largely unchanged since the 1980's. While advancements in neutronics, conjugate heat transfer, and system codes have dramatically improved LWR analysis capabilities, many of these improvement are invalid when applied to the HTGR.

A summary of current VHTR safety assessment tools, modeling capability gaps, and work underway to address those gaps, is provided in "Next Generation Nuclear Plant Methods Technical Program Plan".[14] The safety analysis codes currently suited for gas-cooled reactor applications are "research" codes that have not been endorsed by NRC for regulatory use. These codes were developed independent of the methods qualification guidance contained in RG 1.203 and have undergone significant modification since initial development began in 2000.[23.]

It is recommended that development efforts continue on modular HTGR-compatible design and analysis codes for developmental and regulatory use. A V&V effort may be necessary to confirm them as appropriate for use in regulatory safety calculations. The codes will also provide valuable input to fuel and material testing and test reactor development programs.

<u>RECOMMENDATION 5</u>: Establish plan to systematically develop SFR-compatible safety analysis methods and codes

A brief summary of current SFR safety assessment tools and modeling capability gaps is contained in Chapter 6 of ANL's "Regulatory Technology Development Plan, Sodium Fast Reactor – Mechanistic Source Term Development" report.[7] This documents states that the safety analysis codes suited for fast liquid metal reactor applications are currently "research" codes that have not been reviewed or endorsed by NRC for regulatory use. These codes were very likely developed independent of NRC's methods qualification guidance contained in Regulatory Guide 1.203 and may have undergone significant modifications over ensuing years.

It is recommended that systematic planning be initiated to identify the efforts needed to upgrade promising analysis codes for regulatory use. A dedicated research V&V effort may be necessary to confirm them as appropriate for regulatory safety calculations.

4.3 Core Heat Removal

<u>RECOMMENDATION 6</u>: Continue support to the following modular HTGR-related activities:

6A. Complete experimental tests (prismatic and pebble bed) planned for the High Temperature Test Facility (HTTF)

6B. Complete experimental tests (air-cooled and water-cooled) planned for the Natural Convection Shutdown Heat Removal Test Facility (NSTF)

The ART VHTR TDO currently supports the HTTF at OSU and the NSTF at ANL. The following recommendations encourage continued utilization of these operational facilities.

<u>Recommendation 6A - Complete experimental tests planned for HTTF (prismatic and pebble bed HTGR)</u>

The HTTF facility was constructed at OSU to perform tests that simulate HTGR core behavior undergoing depressurized conduction cool-down with subsequent air ingress. Facility components are configured to replicate prismatic HTGR core conditions but can be altered to reproduce pebble bed core conditions. Additional non-invasive instrumentation is being added to collect high resolution flow and temperature data (under NQA-1 standards) for computational fluid dynamics code validation. This facility represents a major HTGR technology research resource providing data critical to completing a safety evaluation.

Original funding covered 10 initial experiments but approximately 25 are planned.[23] Completing the planned prismatic experiments represent a cost-effective means of addressing important core safety questions and maintains a capability to convert to the pebble bed configuration should that option become necessary.

<u>Recommendation 6B - Complete experimental tests planned for NSTF (air-cooled and water-cooled RCCS)</u>

The NSTF was originally built at ANL to support simulation of the GE-PRISM (SFR) reactor vessel auxiliary cooling system (RVACS). It was refurbished under NGNP to generate data under prototypical HTGR vessel heat-up accident conditions. NQA-1 conformant data has been gathered for the first series of (air-cooled RCCS) experiments in 2014 and continued into 2015. Plans are now being made to convert the facility to support additional water-cooled RCCS experiments.

Information collected from these tests will yield significant insight into the performance of passive vessel heat removal systems for both the HTGR and SFR concepts. It is recommended that these tests continue to be supported to their planned conclusion.

4.4 Additional Recommendations

RECOMMENDATION 7: Form a standing advanced reactor Industry Advisory Group (IAG) with the non-LWR design community

To aid in the challenge of coordinating ART R&D activities with still emergent licensing need, the formation of an informal advanced reactor IAG is recommended. The purpose of the group is to encourage early and frequent dialogue with design vendors concerning ART program development activities. Communications would focus on licensing strategies, technology development challenges, and the exchange of non-proprietary design insights. Industry participation in the ART IAG would be voluntary with membership based on expressed member interest in ART research.

The IAG would complement existing processes and activities, such as the Technical Review Panel (TRP), which relies on a group of experts to review submitted technical information and identify R&D priorities for DOE. Another process focuses on development of Phenomena Identification and Ranking Tables (PIRT) using expert panels (including NRC representatives) to assess safety-relevant phenomena that are then used to define R&D need. While both approaches are effective at preliminarily identifying R&D opportunities, they are generally “episodic” in nature and not conducive to the more frequent interaction forum required for ongoing prelicensing interactions with NRC staff.

An example similar to the proposed IAG approach was the NGNP Licensing Working Group (LWG).[19] This advisory body helped assure DOE-sponsored HTGR technology advancements were systematically linked to the licensing strategies of applicants. Establishing the ART IAG would offer benefits in licensing that include:

- Add greater assurance that the RTDP and ART research priorities are focused on cross-cutting topics and technical challenges that have been identified and confirmed by industry to provide the greatest benefit to future licensing efforts
- Create an environment where industry becomes more familiar with ongoing ART program activities so that the results of ART research can be more systematically utilized in prelicensing interactions with the NRC.

RECOMMENDATION 8: Establish fundamental I&C systems requirements for advanced reactor designs. These requirements will provide insights to establishing future design, fabrication specifications, and testing requirements of necessary I&C equipment

The I&C systems that meet demands of a new power reactor technology and energy market customers must be proven as appropriate, reliable, and robust. From a design perspective, recent advancements in digital I&C may require dedicated testing and research to develop justifications for their use in supervisory and control capabilities at nuclear reactors. Temperature, pressure, flow, and neutron instrumentation may need to operate in higher temperature environments or under corrosive conditions. New combinations of irradiative and corrosive process factors (as might be present in liquid metal fast reactors) could create significant challenges in instrumentation design for which there is no available NRC-endorsed guidance.

It is recommended that specific needs and requirements be established through ART Program and IAG interactions in order to better integrate diverse challenges like these into the R&D goals for I&C development.

5. OTHER CONSIDERATIONS IN ADVANCED REACTOR LICENSING

New reactor concepts are now being discussed that exist at widely different stages of development and design. As a design advance towards maturity, new technology development issues can become significant in the licensing environment. The following subsections identify some major topics that can be expected to become important in future licensing activities for various advanced reactor technologies.

5.1 Regulatory Framework

Multiple approaches have historically been employed to adapt LWR licensing requirements to non-LWR designs. These approaches generally relied upon variations of the LWR-based deterministic safety review where plant operations and accident events are conservatively bounded by a relatively static analysis methodology. The Fort St. Vrain plant (a HTGR located in Colorado) was licensed in the late 1960s and early 1970s using a traditional deterministic approach adapted from LWR requirements which existed at that time.

The attributes of new licensing approaches can vary considerably and, in the case of a newer reactor technology, may encounter considerable challenge with respect to available probabilistic system information. The NGNP project approached this issue by progressively emphasizing the use of risk insights and PRA techniques when adapting deterministic requirements to modular HTGR technology.[19] Using available HTGR design information and precedents set by historic HTGR licensing experience, NGNP developed an approach that emphasized:

- Early identification and resolution (where possible, given outstanding uncertainties in design) of significant policy, technical, and licensing issues relevant to development of a license application for HTGR technology
- Identification and performance of necessary research that supported design advancement and development of a license application
- Engagement of reactor designers, prospective applicants, industry, DOE, NRC, and other stakeholders in pre-application interactions to reduce licensing uncertainties and ascertain the reasonableness of proposed safety approaches
- Delivery of technology familiarization training to NRC staff
- Regulatory evaluation and acceptance concerning the integration of risk insights into the licensing process that were coupled with deterministic engineering judgments to create a risk-informed, performance-based regulatory framework.

The prelicensing goal of NGNP was to establish a regulatory framework amenable to the simplified, inherent and passive safety features associated with a modular HTGR design. A licensing plan was written to guide the interface between R&D required for new HTGR safety systems and concurrent efforts to adapt the existing regulatory structure.[24] A chronology of prelicensing interactions and result are documented in the NRC Licensing Status Summary Report for NGNP.[19]

Experience in NGNP prelicensing suggested that other advanced reactor designs will require similar efforts to adapt the current regulatory framework and ensure appropriate technology-specific risk-informed, performance-based insights can be employed. Formal licensing plans should guide the interface between regulatory framework adaptation and the licensing strategies of applicants. Technology R&D will likely be required to sustain this effort.

The 2012 NRC "Report to Congress: Advanced Reactor Licensing," noted a need for refined regulatory guidance pertaining specifically to the advanced non-LWR design criteria contained in Appendix A of 10 CFR 50. In response, DOE and NRC established a joint initiative to develop Advanced Reactor Design Criteria (ARDC) as a "first step" in developing the framework to aid advanced reactor designers and license applicants. The initial phase of the initiative has recently been completed.[22] The NRC staff is now reviewing proposed ARDCs in the second phase of this two-phase effort.

The NRC Standard Review Plan NUREG-0800 sets[25] forth principle requirements and criteria the staff uses to evaluate nuclear plant safety for LWRs. This review guidance will require adaptation to address non-LWR plant features. The remainder of this Section identifies specific issues that are currently expected to emerge as potential research topics that support NUREG-0800 adaptation.

5.2 Core Disruptive Accidents

A SFR "core disruptive accident" (CDA) can be defined as a highly unlikely event so severe that the reactor core or, more specifically the fuel geometry, is significantly modified over a substantial region of the core. This event can occur if there is either a failure to remove heat from the fuel at a sufficient rate to ensure fuel integrity or there is a local failure in a fuel assembly that propagates beyond that assembly to adjacent regions of the core. Because a CDA can lead to potentially severe consequences and major releases of radioisotopes into containment, extremely conservative safety criteria and associated safety features are typically included in SFR technology development and design efforts. These efforts are intended to make the likelihood of a postulated CDA event sufficiently low that it can be excluded from the plant's design basis.

It is noted that initial studies conducted in the 1970s and 1980s associated with the cancelled Clinch River Breeder Reactor project led NRC staff to (preliminarily) conclude that CDAs need not be included in the design basis for the Clinch River Breeder Reactor.[26]

Discussions with prospective SFR designers should be initiated to determine their current design approach for addressing postulated CDA events. If it is confirmed that applicants will need assistance in addressing CDAs in the standard SFR design, a risk-informed, performance-based research approach should be established to demonstrate how plant safety goals can be confirmed.

5.3 Radionuclide Transport Effects from the Metallic Sodium Pool

A unique feature of the pool-type sodium fast reactor design 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 capability to significantly influence the transport, retention, and scrubbing of radionuclides that might be released from fuel. However, a complete understanding of these phenomena for radionuclides of interest to an MST analysis remains to be fully developed through experimentation.

A recent summary of the radionuclide transport processes associated with these phenomena, and the known information gaps associated with radionuclide transport behavior in a sodium pool, can be found in ANL-ART-3, "Regulatory Technology Development Plan, Sodium Fast Reactor – Mechanistic Source Term Development".[7] Information can also be found in SAND2012-4260, "Sodium Fast Reactor Safety and Licensing Research Plan – Volume I". [5] While extensive sodium metal–radionuclide effects research can be conducted in an ex-core environment, attention must also be extended to the actual basis of in-core radionuclide generation and related transport phenomena under all operational and design accident conditions.

The need for testing concerning the dynamic effects of metallic sodium on radionuclide transport is indicated as a possible research priority but must be confirmed with prospective applicants. If testing is initiated, this research should be scoped in conjunction with fuel test programs, research that supports development of the overall SFR MST, and the safety approaches established by prospective applicants.

5.4 Sodium Technology and Sodium Fire Analysis

The operational history of previous SFR plants suggest that metallic sodium leaks will occur during normal operations. Metallic sodium is highly reactive, potentially corrosive and, in the presence of incompatible constituents like water, generates heat and reaction product hazards that must be evaluated to address their impacts on the design and licensing basis.

In addition, questions also exist about the reliability of components that contact sodium like electromagnetic pumps. Instrumentation will be needed to operate in opaque environments and withstand the corrosive effects of sodium.

5.5 SFR Mechanistic Source Term Development

As stated by NRC in SECY-93-092, a mechanistic source term is established from the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is 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 MST has far-reaching implications for all types of advanced reactors since it establishes postulated effects of the reactor facility on the public and surrounding environment. The NRC has consistently indicated that a mechanistic-type of source term is appropriate for the evaluation of advanced reactor designs.[27, 28]

In 2010, an MST approach for the modular HTGR technology was presented to NRC staff.[20] This proposal established the modular HTGR source term by defining the quantities of radionuclides released from the reactor building to the environment during a spectrum of LBEs. Source terms would be event specific and determined using radionuclide generation and transport models that account for fuel and reactor design characteristics, passive features, and the function of radionuclide release barriers. Since NRC requires sufficient test data to provide adequate confidence in a mechanistic approach, a fuel development and qualification program is to provide the data necessary to better understand fuel performance and fission product behavior. While NRC staff would not approve the proposed approach without submission of a specific reactor design, the strategies, definition, and R&D plans associated with the proposed MST approach was generally determined to be “reasonable”.[16]

ANL-ART-3, “Regulatory Technology Development Plan, Sodium Fast Reactor - Mechanistic Source Term Development”, describes major uncertainties that exist in the MST approach for SFR technology.[7] A foremost uncertainty involves data on fuel performance. Even though DOE has conducted many SFR metal fuel failure tests in the past, these tests were limited to transient overpower failures; it is not known whether the relatively limited scope of these tests will be sufficient to satisfy current MST licensing requirements. It was also noted that there are no current research projects related to characterization of fuel and fission product transport during severe accident events.

The topic of MST development is extremely important to SFR licensing success. Given the technical complexity and long lead times needed to establish a robust technical justification that supports licensing on this topic, interaction with IAG members and NRC staff becomes essential. Limitations in fast neutron irradiation test capabilities further suggest that MST development must be approached in a well thought out and comprehensive manner. Additional information that clarifies the technical options and initial licensing strategy should be gathered and assessed to help identify and prioritize necessary R&D initiatives concerning MST development.

5.6 Deployment of Research, Test, or Prototype Reactor Designs

Many of the activities identified in the RTDP address specific constituents or "separate effects" of a larger set of more comprehensive performance requirements. This may compel reactor developers and applicants to consider some combination of research reactor, test reactor, and prototype plant development as a licensing strategy option to gather integrated information.

A research or test reactor of limited thermal power output may be chosen as a part of the overall strategy to further evaluate, test, and confirm certain features or attributes of a particular reactor technology. The information collected from a highly instrumented first-of-a-kind configuration can provide information essential to license follow-on plants. Results from those efforts may later be combined or "scaled up" to a higher output prototype plant that more directly represents the planned commercial reactor plant offering. Guidance concerning issues that that must be considered and addressed during the development and licensing of a research or test reactor are contained in the NRC's NUREG-1537.[29]

A reactor prototype may be chosen instead of the test/research reactor approach. However, the associated portion of existing NRC regulations (contained in 10 CFR 50.43[e]) provide only high-level requirements with no technology-specific rules or guidance that can be used to assist applicants through the licensing process.

Establishing some combination of advanced non-LWR research, test, or prototype reactors in the future may be essential to enable a comprehensive and integrated advanced reactor technology development capability. If it is determined that this type of licensing approach is necessary to collect essential data, a technology development plan should be formulated that clearly outlines how such an approach can be efficiently utilized and costs, schedules, and licensing risk are minimized while the spectrum of generated test data is maximized. Interactions with NRC staff during plan development will be essential.

5.7 Seismic Analysis of Deeply Embedded Structures

Some advanced reactor designs call for the reactor core and associated heat exchange system to be constructed partially or completely below earthen grade. Up to now, analysis of seismic events during a plant safety review presumed key facility SSCs to be located relatively near the soil surface. Modular HTGR is one class of advanced reactor design that proposes the use of deep embedments to help ensure safe and secure plant operations.

Analysis of seismic events for reactor systems requires that the SSI effects for deeply embedded structures be examined using V&V methods and tools capable of recognizing that influence on analytically predicted seismic responses. These analysis methods and tools are also necessary for the design and qualification of seismic isolators. There is no confirmed capability to perform such an analysis at a nuclear plant using a deep embedment. New methods and tools must be developed that integrate the seismic, structural, and systems-analysis necessary to address deeply embedded and buried structures.

Developing this particular seismic analysis capability is not yet considered a critical licensing concern for nuclear plants intending to use deep embedments. Since some small modular LWR plants are proposing to employ subsurface embedments (i.e., integrated pressurized water reactor, iPWR), development of this seismic analysis capability may already be underway outside the purview of ART. In any event, an NRC-endorsed capability for seismic analysis must be available prior to licensing any reactor that employs a deep embedment.

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