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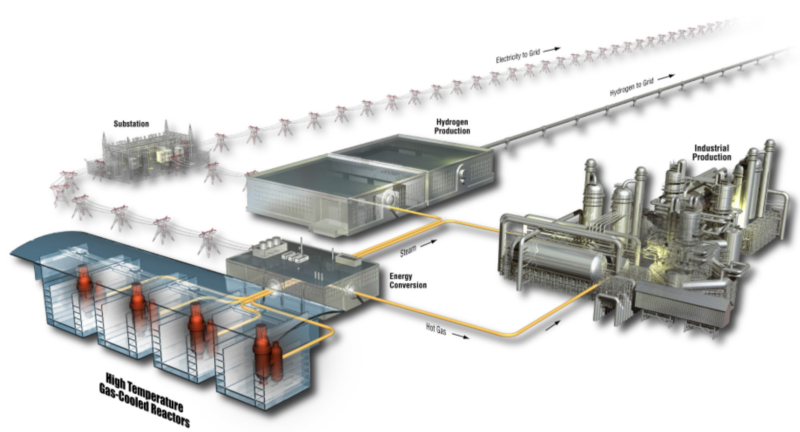
Revision 0

IAEA Cooperative Research Project (CRP) Status Report

James C. Kinsey

September 2017



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IAEA Cooperative Research Project (CRP) Status Report

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Revision 0

September 2017

Approved by:

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Jim C. Kinsey INL ART Licensing Director	Date
	9/5/17
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Summary

This report provides an end-of-year summary that reflects the progress and status of United States (U.S.) activities supporting the International Atomic Energy Agency Cooperative Research Project (CRP) effort to develop modular High-Temperature Gas-Cooled Reactor (mHTGR) Safety Design Criteria. These U.S. activities are being managed by Idaho National Laboratory on behalf of the U.S. Department of Energy, and it is noted that this summary reflects progress during the second year of this planned 3-year activity.

U.S. contributions to this CRP effort in Fiscal Year 2017 are reflected in Appendixes A, B, and C, and include: (1) a summary description of the MHTGR's approach to defense-in-depth; (2) a summary discussion of how the implementation of the proposed safety design process addresses the key lessons learned from the events at Fukushima; and (3) a brief summary of MHTGR research and development gaps and analysis methods needs.

CONTENTS

ACRONYMS

CRP	Cooperative Research Project
FY	fiscal year
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
LBE	licensing basis event
LWR	light-water reactor
mHTGR	modular high-temperature gas-cooled reactor
MST	mechanistic source term
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment
RCM	Research Coordination Meeting
RMRF	Risk Management Regulatory Framework
TECDOC	technical documents
TLRC	top-level regulatory criteria
U.S.	United States

IAEA Cooperative Research Project (CRP) Status Report

1. PURPOSE

This report provides an end-of-year summary that reflects the progress and status of United States (U.S.) activities supporting the International Atomic Energy Agency (IAEA) Cooperative Research Project (CRP) effort to develop modular High-Temperature Gas-Cooled Reactor (mHTGR) Safety Design Criteria. These U.S. activities are being managed by Idaho National Laboratory on behalf of the U.S. Department of Energy, and it is noted that this summary reflects progress during the second year of this planned 3-year activity.

This status summary addresses the remaining CRP deliverables identified in IAEA Research Agreement No. 18737/R0.

2. OBJECTIVES

Modular HTGRs have a number of intrinsic and inherent properties or characteristics that preclude or minimize the potential for large radionuclide release from multi-reactor plant sites to the public. The development and implementation of comprehensive safety design criteria provide a high level of assurance that mHTGRs are consistently designed, constructed, and operated in a manner that takes advantage of these intrinsic properties, while also avoiding unintended compromises in plant safety.

The safety design criteria being developed within this CRP include consideration of an mHTGR adaptation of IAEA's previously-developed light-water reactor (LWR) safety standards (e.g., IAEA SSR-2/1) as one input, while also considering the insights and inputs of member states with experience in mHTGR research and technology development. The criteria developed through this CRP can then be used during the further development of existing and planned HTGRs worldwide to assure that an acceptably broad spectrum of design-basis and beyond-design-basis events are addressed in the designs. The events from the Fukushima-Daiichi accident are being considered during the establishment of the process for developing these criteria.

3. PLANNED CRP OUTPUT DOCUMENTS

The results of the CRP will be documented in an IAEA technical documents (TECDOC) format. Those safety design criteria outputs and the process flow steps developed to establish them, will be considered for inclusion in the future separate activities of the development of IAEA Safety Standards for HTGRs with the cooperation of the department of Nuclear Safety and Security at IAEA.

4. SUMMARY OF U.S. CONTRIBUTIONS IN FISCAL YEAR 2017

Work performed by U.S. CRP members in Fiscal Year (FY) 2017 included efforts and interactions with other members to compare and contrast a risk-informed approach to safety design criteria development ("Approach 1" – largely developed by the United States in FY2016) versus a more prescriptive and deterministic method being proposed by other CRP member states, starting from historical LWR-based criteria ("Approach 2"). The purpose of these CRP members' interactions in advance of the June 2017 Research Coordination Meeting (RCM) was to develop a clear understanding of the strengths and weaknesses of each approach and, if possible, to propose a blended approach to development of mHTGR Safety Design Criteria. "Approach 1" has been developed directly from the process flow utilized by the Department of Energy's Next Generation Nuclear Plant Project for the mHTGR, including insights from the related interactions and reviews provided by the U.S. Nuclear Regulatory Commission (NRC). Approach 1 is also being more fully informed by a currently ongoing U.S. industry-led effort, including regular NRC interactions, pursuing the adaptation of this approach for all advanced non-LWRs through the Licensing Modernization Project. In contrast, "Approach 2" involves a requirement-by-requirement modification to the IAEA LWR Top Requirements. This approach has been used by the Generation IV International Forum for the Sodium Fast Reactor.

In preparation for the June 2017 RCM, all participants developed summary comparisons of the two approaches to provide input from their country's mHTGR program, including the following:

1. The top level requirements in terms of off-normal event frequency and consequences
2. The licensing basis events (Anticipated Operational Occurrences, the Design Basis Events and Accidents, and Beyond Design Basis Events) that are compared to the top level requirements
3. The safety functions required to meet the top level requirements during the events.

The dialogue during the RCM revealed that, although the scope of the two approaches varies significantly, the detailed comparison of the two approaches indicate they're generally consistent for the 15 most critical safety design criteria common in the two approaches.

In addition, the three appendixes to this report contain U.S. deliverables committed to the CRP for FY2017. Those deliverables include:

- A summary description of the mHTGR's approach to defense-in-depth through the incorporation of multiple independent barriers to radiouclide release. This description includes a summary of available U.S. regulator's assessment of this approach to defense-in-depth.
- A summary discussion of how the implementation of the proposed safety design process addresses the key lessons learned from the events at Fukushima (e.g., events impacting more than one reactor on a multi-reactor site, interdependence of radionuclide release barriers, reliance on active systems and operator actions, and adequate cooling of spent fuel)
- A summary of research and development gaps and analysis method verification and validation needs in support of the safety case for the mHTGR.

5. PLANNED U.S. CONTRIBUTIONS FOR FISCAL YEAR 2018

Planned U.S. contributions to this CRP effort in FY2018 include continuing the development of assigned sections of the IAEA TECDOC, which commenced in FY2017, to reflect safety design criteria for mHTGRs using an integration of Approaches 1 and 2. Members of the U.S. team will also be reviewing TECDOC inputs from other CRP members as a part of that integration effort. It is anticipated that all TECDOC inputs will be completed, and the document finalized, in calendar year 2018.

6. SUMMARY OF CRP CHALLENGES

The primary and continuing challenge for the CRP is to establish an agreed upon process flow that allows for the integration or "blending" of the risk-informed performance based "Approach 1" with the largely LWR-based and prescriptive "Approach 2". This integration is further challenged by the differing regulatory structures and regulator expectations in the various member states, and the LWR-based precedents established within existing IAEA Safety Standards. There was extensive dialogue addressing this topic through FY2017 and during the June 2017 RCM, with a planned path forward now being implemented by CRP members in parallel with the development of the draft content of the IAEA TECDOC under development.

Appendix A – Summary Description of the mHTGR Approach to Defense-in-Depth

This appendix provides a summary description of the modular high-temperature gas-cooled reactor's (MHTGR's) approach to defense-in-depth through the incorporation of multiple independent barriers to radionuclide release. The summary includes a discussion of available U.S. regulators' assessment of this approach to defense-in-depth.

Defense-in-depth is a safety philosophy in which multiple lines of defense and conservative design and evaluation methods are applied to ensure the safety of the public. The philosophy is also intended to deliver a design that is tolerant to uncertainties in knowledge of plant behavior, component reliability, or operator performance that might compromise safety. This appendix includes a review of the regulatory foundation for defense-in-depth, a definition of defense-in-depth that is appropriate for advanced reactor designs based on high-temperature gas-cool reactor (HTGR) technology, and an explanation of how this safety philosophy was viewed for use by the Next Generation Nuclear Plant (NGNP) Project.

The term "defense-in-depth" is used sparingly in U.S. Nuclear Regulatory Commission (NRC) requirements, but is generally stated as a "philosophy" or a "concept," and those requirements are stated simply as, "Defense-in-depth shall be maintained." Guidance in the NRC Standard Review Plan provides further definition:

> *"Defense in depth is defined as a philosophy that ensures that successive measures are incorporated into the design and operating practices for nuclear plants to compensate for potential failures in protection and safety measures. In risk-informed regulation, the intent is to ensure that the defense-in-depth philosophy is maintained, not to prevent changes in the way defense in depth is achieved. The defense-in-depth philosophy has been and continues to be an effective way to account for uncertainties in equipment and human performance."*

Based on the analysis of NRC historical literature, requirements, guidance, and policy papers, and by considering the principles described by the International Atomic Energy Agency, it is proposed that the NGNP framework for defense-in-depth address the three major elements summarized below and illustrated in Figure A-1:

- **Plant Capability Defense-in-Depth** reflects the decisions made by the designer in the selection of functions, structures, systems, and components for the design that ensure defense-in-depth in the physical plant.
- **Programmatic Defense-in-Depth** reflects the decisions made regarding the processes of manufacturing, constructing, operating, maintaining, testing, and inspecting the plant and the processes undertaken that ensure plant safety throughout the lifetime of the plant.
- **Risk-Informed Evaluation** of defense-in-depth reflects the development and evaluation of strategies that manage the risks of accidents, including the strategies of accident prevention and mitigation. This aspect of defense-in-depth also provides the framework for performing deterministic and probabilistic safety evaluations, which help determine how well various Plant Capability Defense-in-Depth and Programmatic Defense-in-Depth strategies have been implemented.

Each of these elements of defense-in-depth is supported by a comprehensive probabilistic risk assessment (PRA) and a parallel set of deterministic evaluations that are performed to ensure that all decision making in these processes is systematically evaluated in a comprehensive risk-informed manner. The PRA is based on plant design, extensive deterministic bases, and a specification of the capabilities of the plant SSCs in the performance of their functions, including the plant safety functions. The results of the PRA expose the characteristics of the Plant Capability Defense-in-Depth and are dependent on the safety margin and reliability of each SSC modeled in the PRA.

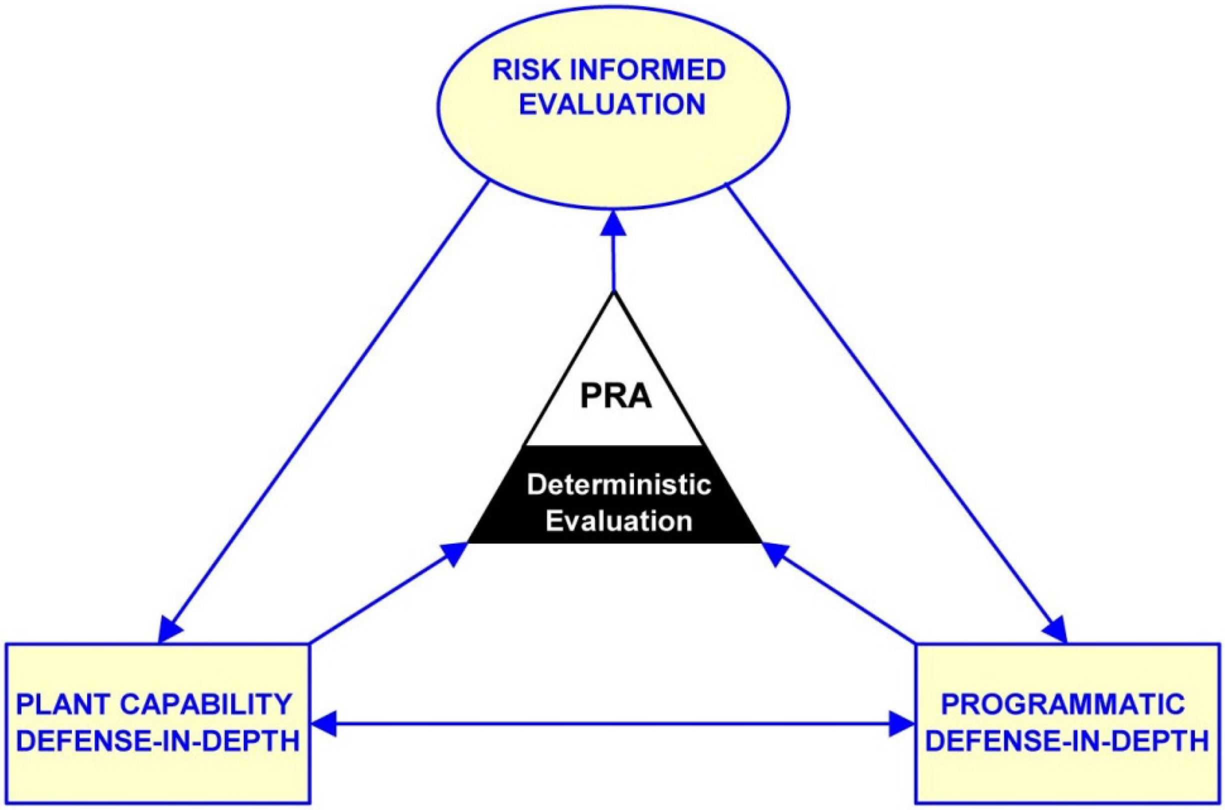

Figure A-1. Major Elements of Defense-In-Depth Framework.

This mHTGR defense-in-depth approach embraces the engineering and regulatory practices that have evolved over the last 50 years of reactor design and licensing and more completely integrates risk-informed, performance-based methods into the plant design process. This process combines deterministic and probabilistic methods into a robust fabric designed to expose relationships in design and operation.

The proposed methodology also provides for compensatory special treatment requirements in design, manufacturing, construction, testing, operations, and maintenance to compensate for uncertainties in the design and analysis process, thus providing a high confidence that equipment will perform as expected. The result will be a set of conservative design features combined with inherent reactor characteristics, passive design features, and active systems to: (1) prevent transients and accidents; (2) ensure the performance of safety functions; (3) prevent the release of radioactive material; and (4) mitigate the consequences of accidents.

The principles of multiple, independent, and concentric barriers to radionuclide transport are assessed for each significant source of radioactive material to ensure that defense-in-depth has been maintained. In addition, the principles of design margin, redundancy, and diversity would be applied in the design of the structures, systems, and components that support the required safety functions and maintain the integrity and effectiveness of these barriers.

These defense-in-depth strategies ensure that the top-level regulatory criteria are met, adequate safety margins are demonstrated, deterministic principles of defense-in-depth are included, and that residual uncertainties in the reliabilities and capabilities of the structures, systems, and components (SSCs) providing the required safety functions are adequately addressed for the life of the plant.

NRC concluded in its review that this approach is largely consistent with the approach that the U.S. Department of Energy proposed for the MHTGR in the mid-1980's and that the staff evaluated as described in its draft MHTGR Pre-application Safety Evaluation Report (NUREG-1338), which was initially issued in 1989 and then updated in 1995. However, NRC made reference to other agency activities related to the topic of defense-in-depth that were underway in 2014 when this NGNP-related assessment was completed. NRC noted that the proposed NGNP approaches to defense-in-depth are conceptually similar to those that were being considered in NRC work related to NUREG-1860 and NUREG-2150, which outline a Risk Management Regulatory Framework (RMRF), and therefore deferred its feedback on the defense-in-depth approach proposed by NGNP.

The NRC staff then completed its assessment of the RMRF approach suggested by NUREG -2150, and concluded that this approach should not be implemented for currently operating nuclear power reactors. However, they also summarized their belief that the adoption of a risk-informed regulatory framework, similar in concept to an RMRF, would provide the greatest benefits for new reactor designs that employ non-traditional technologies (e.g., Generation IV designs), and indicated that the NRC staff would continue to engage stakeholders interested in pursuing such a risk-informed framework. This follow-on industry stakeholder engagement is now underway through the Licensing Modernization Project, which is an industry-led effort being supported by the Department of Energy and coordinated through the Nuclear Energy Institute to establish a risk-informed and performance-based approach for the development, licensing, and operation of advanced non-light-water reactor (Generation IV) design. Licensing Modernization Project progress is being monitored, and associated results will be incorporated, where appropriate and agreed upon by International Atomic Energy Agency Cooperative Research Project members, into the HTGR safety design criteria technical documents being developed.

Appendix B – Summary Discussion: Addressing Fukushima Lessons Learned

This appendix provides a summary discussion of how the implementation of the proposed safety design process addresses the key lessons learned from the events at Fukushima.

A primary focus of the June 2017 Research Coordination Meeting was to compare and contrast the performance-based and risk-informed approach to safety design criteria development proposed by the United States ("Approach 1") versus a more prescriptive and deterministic method, starting from historical light-water-reactor-based criteria ("Approach 2") to develop a clear understanding of the strengths and weaknesses of each and, if possible, to propose a blended approach to development of mHTGR safety design criteria. "Approach 1" has been largely developed directly from the process flow utilized by the United States Department of Energy's Next Generation Nuclear Plant Project for mHTGR designs, including insights from the related dialogue and reviews provided by the U.S. Nuclear Regulatory Commission.

"Approach 1" proposed by the U.S. is characterized by two key concepts that relate directly to more clearly understanding and addressing the lessons learned from the events at Fukushima. Those two concepts include: 1) the use of a risk-informed and performance-based process for identifying the spectrum of licensing basis events (LBEs) to be evaluated and addressed by the design; and 2) the use of a mechanistic source term (MST) approach when evaluating the potential radiological releases that may occur from those events. These two concepts are summarized below.

Identification and Evaluation of LBEs

LBEs are a comprehensive set of event sequences used in development of the license application that form the basis for plant analysis and represent the plant's characteristic performance in all analyzed frequency and consequence ranges and modes of operation.

Approach 1 is a systematic, performance-based and risk-informed methodology for selecting and classifying LBEs for the mHTGR, consistent with current Nuclear Regulatory Commission policies and guidance on the application of deterministic design criteria and the use of probabilistic risk assessment (PRA) techniques. The methodology integrates the use of deterministic safety principles and PRA insights as critical inputs into the selection of LBEs. A combination of deterministic and probabilistic analysis is used to identify these events and evaluate the event sequences. The LBE selection process 1) identifies event sequence families based on an identified set of initiating events, 2) establishes the frequency of each of these event sequences, and 3) assesses the consequences of the event sequences against offsite dose criteria.

LBE selection is also an integral part of the overall design process. The design attributes of the plant influences the type and sequence of events of LBEs, and the initial set of identified LBEs can be used to improve the final design. Once the initial set of LBEs is identified, the design can be refined to reduce the frequency or consequence of a given LBE. This suggests an iterative design process where more design and analysis detail is available at each phase of the design process, and includes the following development sequence:

1. A deterministic approach is used to select an initial event set providing a starting point for a given phase of the design process. For example, a set of initial events developed from conceptual design provides the starting point for preliminary design.
2. The LBEs are updated as the design and analysis progress. The PRA is developed and revised as the design matures. This begins to risk inform the LBE event sequences with insights gained from the design phase PRA.

3. A review of the LBEs is performed at the end of the design phase to evaluate conservatisms in the selected events.

This approach uses a blend of deterministic and probabilistic techniques, where deterministic evaluations provide initial identification of events and subsequent confirmation of margins through the evaluation of DBAs, and probabilistic components provide a systematic process to ensure all sequences are captured and properly classified.

Approach to Establishing MSTs

The mechanistic approach to source term development is needed to establish the technical basis and take credit for the radionuclide retention capabilities of the multiple barriers to radionuclide transport consistent with the mHTGR safety design. For each LBE identified using the process described above, MSTs are developed that evaluate the realistic response of the plant to the initiating event. The initial radionuclide inventories during the modes of normal operation will include, as appropriate those in the fuel, the circulating activity, the plate-out activity within the helium pressure boundary, the spent and used fuel, and radioactive waste systems. For each of these inventories, the response to the initiating event of the barriers and that of the passive and active SSCs that protect those barriers are modeled.

Several factors need to be considered in a mechanistic definition of event-specific source terms for mHTGRs. As these factors are defined and characterized, the influence of each on the calculated dose is established. This permits developing a target for each element in the source term calculation to meet the safety goals of the project. The development of these targets is addressed by the following process:

1. Establish the top-level radionuclide control requirements to ensure the health and safety of the public and plant workers and to protect the environment.
2. Identify LBEs for which plant conditions and source terms are to be calculated and compared with the goals.
3. Identify and characterize the factors affecting radionuclide generation and transport for this reactor technology.
4. Scope the influence of each factor on the magnitude of the source terms and establish the principal parameters needed to characterize the effect of these factors on the generation and transport of radionuclides for the LBEs.
5. Establish a target for each factor to achieve the goal for each event.
6. Calculate source terms and dose rates based on the current understanding of generation and transport phenomena for the LBEs and compare with top-level regulatory criteria (TLRC) requirements.
7. As needed to support meeting the TLRC requirements, identify how well each factor is currently characterized to validate its target in establishing the source terms and, where the current characterization is deficient, define the gaps between what is needed and what is known.
8. Develop and complete analytic and testing programs to fill those gaps, if needed.
9. Recalculate source terms and dose rates again based on the more fully characterized and validated generation and transport phenomena for the LBEs and compare with TLRC requirements.

This MST approach analyzes a mHTGR functional containment comprising several barriers that limit the release of radionuclides to the environment for each postulated event, including normal operating conditions, abnormal operating conditions and accident conditions. The multiple barriers include:

- Individual fuel particle kernels and coatings,
- The fuel matrix and fuel element graphite,

- The helium pressure boundary (primary circuit), and
- A vented low-pressure reactor building.

Design methods for determining radionuclide source terms, which include analytical tools used to calculate the performance of each of these barriers during radionuclide transport under event-specific conditions, are defined and supported by testing and analysis. These analytical tools are applied in calculations for normal operating conditions, abnormal operating conditions, design-basis accident conditions, and beyond-design-basis-accident conditions.

Summary of How Approach 1 addresses some of the more critical causes of the events at Fukushima Daiichi

1. Events impacting more than one reactor on a multi-reactor site

   Proposed Approach 1 systematically evaluates all identified event sequences and event sequence families on a "per plant-year" basis. This means that all reactor units on a plant site are evaluated both individually, and in combination, for event sequences that may apply to multiple reactors simultaneously. This assures that the overall impact of the plant facility is evaluated and considered, rather than evaluating impacts to the public based on single reactor units only, as was the case at Fukushima Daiichi.

2. Interdependence of radionuclide release barriers

   The developers of mHTGRs typically design and configure the various release barriers to be largely independent. Approach 1's source term approach, combined with the LBE determination process, provides a tool for evaluation of those design choices with any related interdependencies, so that the adequacy of release barrier configurations can be confirmed, including considerations of defense-in-depth.

3. Reliance on active systems and operator actions

   The mHTGRs being addressed by the safety design criteria under development within this Cooperative Research Project typically use the inherent high-temperature characteristics of tristructural-isotropic-coated fuel particles, graphite moderator, and helium coolant, as well as passive heat removal from a low-power density core with a relatively large height-to-diameter ratio within an uninsulated steel reactor vessel. These mHTGRs are designed in such a way to ensure during design-basis events (including loss of forced cooling or loss of helium pressure conditions) that radionuclides are retained at their source in the fuel. They generally do not rely on either active safety systems or operator actions to achieve these capabilities. However, this aspect of a particular design is evaluated and confirmed through the Approach 1 process.

4. Adequate cooling of spent fuel

   Adequacy of spent fuel cooling, as well as all other critical cooling functions of any given mHTGR concept, are directly addressed through the event evaluation process summarized above. This includes a review of available heat removal systems and configurations to assure margins and capabilities.

Appendix C – Identification of Significant Research & Development Gaps and Analysis Method Verification and Validation Needs

The key research and development related activities that need to be considered and addressed in support of defining and confirming the safety case for the modular high-temperature gas-cool reactor were previously evaluated and reflected in a U.S. Department of Energy-sponsored activity within the Advanced Reactor Technology Program. PLN-4910, "ART Program Regulatory Technology Development Plan," was originally issued in 2015. It is currently undergoing revision, which will be issued in the very near term. PLN-4910 should be referenced for a more comprehensive description of significant and higher priority research and development gaps or verification and validation needs, although the needs in those areas are briefly summarized below.

Fuel Qualification and Mechanistic Source Term

- Establish fuel service conditions and performance requirements for normal and off-normal operations
- Demonstrate fuel performance requirements are met at normal operating conditions using irradiated fuel at design conditions, fuel irradiation performance monitoring, and post-irradiation examinations
- Demonstrate fuel performance requirements are met for accident conditions using irradiated fuel at accident conditions and monitored fuel accident performance
- Establish and validate models for fuel performance and radionuclide transport in fuel
- Develop fuel product and fuel fabrication process specifications
- Develop event-specific mechanistic source terms based on fuel test program results
- Establish and validate models for radionuclide transport to the environment
- Demonstrate mechanistic source term models in best estimate and conservative analyses of transients and accidents.

Core Heat Removal

- Confirm reactor core heat removal capabilities.

Materials Analysis

- Irradiation and property testing of advanced reactor materials and application development
- Development of material codes and standards.

Analytical Codes and Methods

- Define calculational envelope required to analyze reactor systems
- Define evaluation models capable of an analysis across the calculational envelope defined by the above
- Identify data or perform thermal fluid experiments to generate comprehensive database for validating design safety evaluation models.