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Advanced Nuclear Technology: Evaluation of Risk Analysis Methods and Tools for Advanced Reactors

Advanced Nuclear Technology: Evaluation of Risk Analysis Methods and Tools for Advanced Reactors

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ABSTRACT

Risk analysis, particularly probabilistic risk assessment, is a vital element in the development, licensing, and deployment of the next generation of nuclear power plants. Advanced reactors (ARs), however, will operate differently than the existing light water reactors for which risk methods and tools have been developed and successfully implemented. For example, many of the emerging AR designs will depend on advanced nuclear fuels and passive safety systems. Such differences present opportunities to apply risk analysis in a new and optimized way.

EPRI is reviewing current risk analysis methods and tools to assess their readiness to support AR designs, and this report develops plans for further research and development to ensure these methods and tools can support ARs through design, licensing, construction, and operation. EPRI is building on its experience supporting the current nuclear fleet, thereby ensuring a solid foundation for providing solutions to AR developers and end users around the world. This evaluation results in a short list of higher-priority topics for short-term research, including improvements to risk metrics, passive safety system reliability, external hazards, human reliability, and data.

Keywords

Advanced reactors
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KEY RESEARCH QUESTION

Risk analysis, particularly probabilistic risk assessment, is a vital element in the development, licensing, and deployment of the next generation of nuclear power plants. Advanced reactors, however, will operate differently than the existing light water reactors for which risk methods and tools have been developed and successfully implemented. This project addresses questions related to the readiness of current risk analysis methods and tools for supporting advanced reactors and identifies the technical gaps for upcoming research and development.

RESEARCH OVERVIEW

To evaluate the readiness of risk analysis methods and tools for advanced reactors, this project performed several coordinated activities. A literature search clarified the broad range of issues advanced reactors are facing, the current state of the art in risk analysis, and the areas of ongoing research on these issues. Interviews with several advanced reactor developers and current nuclear utility organizations provided insights on issues they are facing related to risk analysis. Finally, an EPRI workshop brought together stakeholders from advanced reactor developers, current nuclear utility organizations, industry organizations, national laboratories, international organizations, government regulators, industry consultants, and academia to present and discuss those risk analysis topics on which research is necessary to support the development and deployment of advanced reactors. This report consolidates insights from these activities and evaluates the priorities for the most commonly discussed issues.

KEY FINDINGS

- Key research needs fall into the following categories:
  - Methods to assess system reliability and overall risk impacts, including impacts of external events
  - Methods to assess human reliability and its impact on advanced reactor safety
  - Methods to evaluate long-duration events and potentially unique end states
  - Methods to address risk using a broader and more integrated approach (e.g., enterprise risk)
  - Methods to address plant and component data issues
- The research topics with the highest priority for short-term investigation are:
  - Developing a generically applicable approach to evaluate passive system reliability
  - Identifying appropriate risk metrics for advanced reactors
  - Assessing the impact of very low frequency external hazards, such as seismic events
  - Evaluating new human reliability analysis issues, such as errors of commission
  - Identifying and prioritizing data needs for plant safety analysis and risk assessment

WHY THIS MATTERS

This research identifies the key technical gaps in risk analysis methods and tools that need to be addressed to support advanced reactor design, licensing, construction, and operation. These issues provide a roadmap to guide industry research in this area and ensure the readiness of risk analysis methods and tools for the advanced reactor community.

HOW TO APPLY RESULTS

EPRI will use the results of this report to guide its near-term research on risk analysis methods and tools for advanced reactors and is strongly interested in ongoing collaboration with members. This collaboration can consist of continued information sharing as well as more in-depth partnerships with members on pilot studies.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- EPRI maintains public- and member-facing advisory groups under the Advanced Nuclear Technology (ANT) Program that focus on advanced reactor R&D, demonstration, and commercialization topics. These forums provide opportunities to exchange information and obtain input on the direction and nature of EPRI's ANT programmatic focus to support deployment of advanced reactors.
- EPRI continues to look for and welcome collaborative opportunities to develop and apply tools and methods that support commercialization of advanced nuclear technology.

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PROGRAMS: Nuclear Power, P41; Advanced Nuclear Technology, P41.08.01; and Risk and Safety Management, P41.07.01

IMPLEMENTATION CATEGORY: Reference–Early R&D

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1 INTRODUCTION

1.1 Historical Background in the United States

The role of probabilistic risk assessment (PRA) in the licensing and operation of nuclear power plants (NPPs) has changed significantly over the past 40 years. Because PRA technology was not developed and applied to assess NPP safety until performance of the WASH-1400 study in 1975 [1], the first generation of NPPs was licensed and operated for many years without the use or benefits of PRA models or the insights derived from such models. For plants licensed in the United States during the 1970s and 1980s, there was no requirement to develop a PRA as part of the licensing basis. Based on the insights obtained from PRA models and the benefits obtained from risk-informed decision making and related applications, a PRA model is required as part of risk-informing the licensing basis for new light water reactors (LWRs) under Part 52 of Title 10 of the Code of Federal Regulations [CFR] [2]. For example, a PRA meeting the requirements specified in the 2009 ASME/ANS PRA Standard for LWRs [3] was required to support the Vogtle 3&4 combined operating license, fuel loading, and startup. PRAs also are required for licensing of light water small modular reactors (SMRs), and it is expected that PRAs also may be required for most advanced non-LWRs (NLWRs) in the new Part 53 to 10 CFR, which currently is undergoing development.[1]

PRAs have proven to be a powerful tool to complement traditional, deterministic safety analyses throughout the life cycle of an NPP. Application of the technology to support risk-informed decision making at operating NPPs has contributed to improved operational performance of NPPs while simultaneously improving safety. This is shown in Figure 1-1, which displays increasing plant capacity factors as a measure of operational performance with a simultaneous decrease in average estimated plant core damage frequency (CDF) as a measure of safety improvement over the 23-year period 1992–2014, as reported by EPRI [4]. Similarly, the Nuclear Energy Institute (NEI) conducted a comprehensive evaluation of U.S. industry performance that evaluated a range of indicators of both operational and safety performance. The key elements that contribute to the observed improvements in performance in both plant operation and safety across the industry are depicted in Figure 1-2 (taken from NEI 20-04 [5]).

[1] For advanced reactor deployments that will be licensed under 10 CFR 50, Section 34(f)(1)(i) also requires development of a PRA. In particular, this section requires that the license applicant "Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

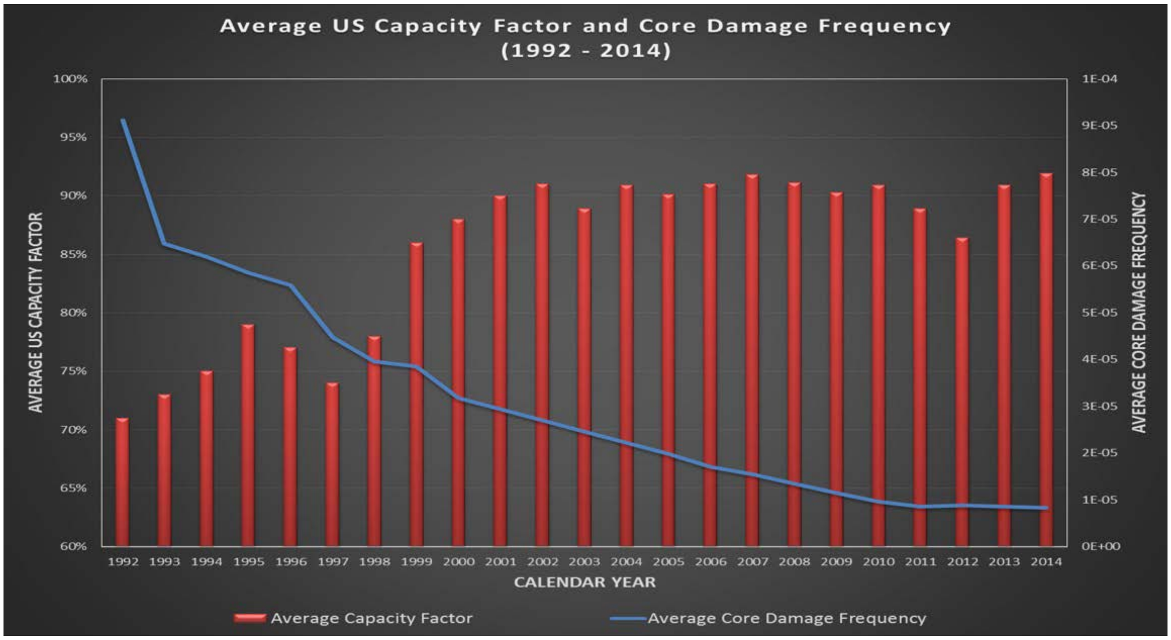

Figure 1-1
U.S. industry trends in plant capacity factor and CDF, 1992–2014 [4]

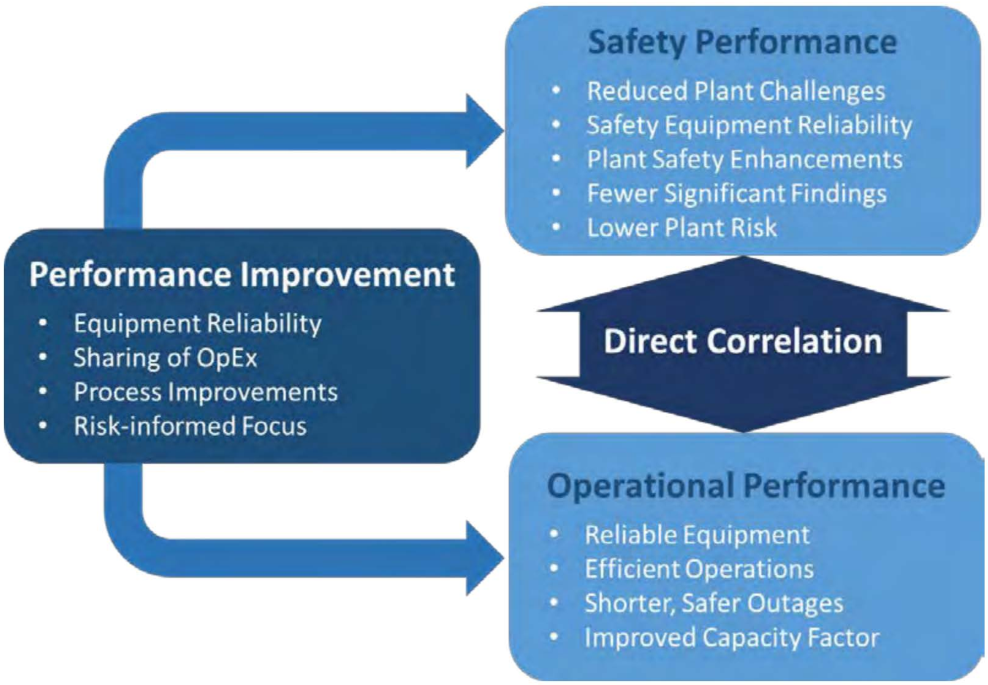

Figure 1-2
Key elements in achievement of high levels of operational and safety performance [5]

Many of the features that classify new reactor designs as "advanced" also present challenges for the approach currently employed to conduct risk assessments of plant safety for the current fleet of LWRs. Many advanced reactors (ARs) are designed to include less inventory of radioactive materials, more stable fuel forms, higher system thermal capacities with longer thermal (time) constants, and passive safety features that rely on natural forces (e.g., gravity). In addition, many ARs employ different materials (e.g., liquid fuels with higher levels of ^{235}U enrichment) and different physical processes (e.g., fast neutron spectrum) than existing LWR NPPs. Further, the number of diverse reactor designs undergoing active development, many of which possess different combinations of advanced features, provides a significant challenge with respect to assessing plant safety and risk. For the purposes of this research, the term *advanced reactor* is defined to include all NLWRs (often referred to as Generation IV reactors), as well as SMRs, which can include LWR moderated and cooled reactors. The research described in this report primarily considers the conventional application of ARs to the generation of electrical power; however, it is anticipated that the research will be applicable and can be adapted to the deployment of ARs for other applications (e.g., district heating, saltwater desalination, or hydrogen production). Specific, unique aspects of such applications may need to be considered in the assessment of risks.

Key Definition

Advanced reactors: Non-light-water reactors and small modular light water reactors

As a consequence of the characteristics of ARs, requirements for developing and applying PRAs have changed. To reflect these changes, a standard that specifies these requirements for developing a PRA for NLWR ARs has been developed [6]. This standard has been reviewed and approved for provisional use to support licensing of NLWR ARs in the United States in Regulatory Guide (Reg Guide) 1.247 [7]. Additionally, because licensing and operation of these ARs represent a first-of-a-kind (FOAK) application, the U.S. nuclear industry developed guidance (NEI 18-04 [8]) for a modern, technology-inclusive, risk-informed, and performance-based (TI-RI-PB) process. This TI-RI-PB approach provides guidance for the selection of licensing basis events (LBEs); safety classification of structures, systems, and components (SSCs) with associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for NLWR ARs. Regulatory review of the NEI 18-04 guidance document determined the approach provides an "acceptable means for addressing … topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection," and Reg Guide 1.233 [9] provided regulatory endorsement for its use in the United States.

A critical challenge for ARs is related to the evolution of PRA and its use for LWRs. By the time PRA became a key aspect of the regulation of LWRs, there had been thousands of reactor years of operational experience (OE) and three decades of data collection, testing, and methods development. The industry is now approaching five decades and thousands more reactor years of experience. This OE has provided a strong basis for the accuracy of inputs to LWR PRAs and substantially reduced the uncertainty in the risk estimates. Given that many advanced NLWR technologies constitute FOAK applications, there are substantial uncertainties in the methods and tools that will be used to analyze and manage risk at these plants. Due to these uncertainties, there also will be challenges in the analysis and interpretation of the results of the risk assessments performed to support their licensing and operation. This situation does not imply too little data or limitations in the methods available to perform a sound and useful PRA for an NLWR plant. Rather, the existing experience has not been mined and organized in a manner conducive to developing PRA models in an efficient, effective, and consistent manner.

The objective of the research described in this report is to identify and prioritize technical issues related to the conduct of risk assessments of advanced nuclear plants, with greater consideration/priority given to identified gaps important to near-term licensing actions for ARs. The outcome is a research roadmap that identifies and prioritizes areas where research is needed to develop, validate, and disseminate the data, methods, and tools needed to better analyze plant risk for AR designs. The execution of the research roadmap is intended to support the planned timelines for the licensing and deployment of the various AR designs undergoing active development.

1.2 Purpose

PRA is a vital element in the licensing of many generations of NPPs. However, ARs employ more advanced systems and are designed to operate differently than the existing operating LWRs for which existing risk methods and tools have been developed and successfully implemented. Accordingly, the current PRA methods and tools need to be assessed for their readiness to support specific AR designs, then further developed to support the licensing and operation of these plants. This report builds on existing EPRI research and resources to assess and develop PRA methods and tools for implementation in ARs. Specifically, it researches and documents the current readiness of existing risk methods and tools to support the analysis, licensing, and operation of ARs. The primary outcome of this research is a roadmap to guide subsequent phases of research.

1.3 Scope

The current state of the art for AR development covers a very wide range of plant types, sizes, and technologies. This section documents the scope of this report by describing the considerations determining what technologies it does and does not include. Many of the methods and tools in this report may still apply to some areas not explicitly included in the scope of the research. For example, many of the methods and tools may be usefully applied to assess risk and safety for sea-based reactors, although these reactors are not explicitly included and will have special considerations that are not addressed herein.

The following provides a broad outline of the elements of ARs considered in this research:

- AR size:

  ARs can range in size from large reactors of thousands of megawatts to small, compact, and portable reactors of tens of megawatts or less (so-called microreactor designs).

  – This report addresses all sizes of nuclear power generation facilities.

- End uses:

  Because of the wide variations in size and performance characteristics, ARs can be used for a variety of applications. As indicated previously, the primary focus of this research is on application of ARs to electrical power generation.

  – This report addresses land-based power generation facilities.

  – It does not address sea-based or space-based reactors (e.g., reactors for NASA) or reactors used for other applications, such as thermal heating, generation of hydrogen, or desalinization of water.

- Core types:

  Coincident with variations in size, ARs can have a variety of core types.

  – This report addresses all core types, including solid fuels made of pellets, balls, or blades; liquid fuels (such as molten salt fuels with liquid cores); solid fuels with liquid coolants (such as solid fuel pellets with liquid metal coolant) or gaseous coolants (such as high-temperature gas-cooled reactors). It includes modular cores and units with multiple modules.

- Fuel types:

  – This report addresses all fuel types, ranging from existing fuels (e.g., conventional ceramic fuel pellets with cladding and tri-structural isotropic particle fuel pellets) to recently developed accident-tolerant fuel designs to nonconventional advanced fuels, including fuel in a molten form.

- Plant system characteristics:

  – This report addresses issues related to technologies used in existing nuclear plants, such as digital instrumentation and controls (I&C), and new technologies, such as passive systems.

  – It does not address materials-related issues that may arise in AR designs.

- Sources of radiation:

  – This report addresses events that can lead to a radiation exposure to the public. These events include widespread core damage in a reactor vessel or in distributed fluid systems associated with use of molten core designs, as well as widespread fuel damage within a spent fuel pool.

  – It does not address operational failures of individual fuel elements, dry cask storage failures, or failures/leakage of radiographic sources; also, events associated with fuel transportation to and from the site of the power plant.

- Types of assessment:

  Additional assessments beyond those needed for nuclear safety may be required or deemed to be beneficial as part of a risk/hazard assessment of an AR.

  – This report addresses PRA of events that result in large-scale degradation of the nuclear fuel that can impact public health and safety; also, other types of risk (e.g., operational, economic).

- It does not address additional hazards such as those from co-purposed applications (e.g., hydrogen generation, district heating, and energy storage devices), safeguards, and security (physical and cyber-security).

- Types of PRA:
  - This report addresses the equivalent of a Level 1, Level 2, and Level 3 PRA[2] for existing reactors that quantifies the frequency of initiating events in all modes and all hazards that can lead to a radioactive release with potential consequences to the public. This includes a discussion on risk-informed decision making using PRA in conjunction with supplemental information, such as evaluations of safety margins and DID.
  - It does not address use of analysis methods not employed in current-generation PRA models (such as dynamic PRA and modeling/simulation-based approaches) or derivatives used for ongoing risk management decision making (such as configuration risk management models).

[2] Level 1 PRA examines events that cause damage to the nuclear reactor fuel; Level 2 PRA examines the transport and release of radionuclides from the facility; Level 3 PRA examines the offsite consequences of the release of those radionuclides.

2

REVIEW OF EXISTING GUIDANCE FOR LICENSING OF ADVANCED REACTORS

Advanced nuclear reactors are viewed as a viable solution for producing electricity and meeting climate change goals, as well as for use in non-electric applications (such as hydrogen production). The designs of advanced NPPs cover various reactor technologies beyond water-cooled reactors, such as high-temperature gas-cooled reactors, molten salt reactors, and liquid metal-cooled fast reactors. ARs (which, as indicated in Section 1, for the purposes of this research include SMR designs that use LWR technology) have several advanced design safety features that are intended to substantially reduce the likelihood and/or consequences of a reactor accident.

Many countries are interested in deploying ARs; however, there is limited operational or regulatory experience with these reactors. For example, only a few nuclear regulatory authorities have issued construction permits or operating licenses for SMRs, and uncertainties exist regarding the best approach to regulating them. In particular, the advanced safety features that respond to the unique hazards and challenges involving the physical barriers in advanced NPP designs can be different from those for water-cooled reactors.

In the International Atomic Energy Agency (IAEA) Advanced Reactors Information System [10], more than 50 SMR designs are identified, covering a variety of reactor coolants, nuclear fuel designs, and neutron spectra. Recently, significant advancements have been made in both reactor design requirements and licensing processes for SMRs, particularly water-cooled reactors such as NuScale, BWRX-300, CAREM-25, and Akademik Lomonosov reactors, as well as high-temperature gas-cooled reactors such as the HTR-PM, X-Energy Xe-100, and GTHTR300 reactors. As a result of these advancements and the need for reliable and sustainable energy sources, successful deployment of ARs is expected to lead to improved performance, economic effectiveness, and a significantly higher level of safety.

2.1 General Guidance on Advanced Reactor Licensing

Because it is recognized that the expanded application of nuclear power technology will be needed to achieve identified reductions in greenhouse gas emissions, governmental bodies and regulatory authorities are actively developing requirements and guidance to support the licensing and regulation of ARs. Examples of such development within the international community include work by the IAEA, the United States Nuclear Regulatory Commission (NRC), and the Canadian Nuclear Safety Commission (CNSC). The status of the requirements and guidance developed through these efforts is summarized in this section.

In the United States, the NRC has developed guidance on how the general design criteria (GDCs) specified in Appendix A of 10 CFR Part 50 [11] may be adapted for the licensing of NLWR designs. This evaluation was published as Reg Guide 1.232 [12]. This Reg Guide provides guidance for developing principal design criteria (PDCs) for NLWR designs. The goal of the Reg Guide is to help designers and applicants understand the NRC's expectations for the content and format of PDCs and to ensure that the PDCs provide reasonable assurance of adequate protection of public health and safety. The scope of PDCs includes safety-related SSCs and operator actions, and the Reg Guide provides guidance on the format and content of PDCs, including the use of performance-based criteria and quantitative risk criteria. A useful aspect of the Reg Guide is a "crosswalk" review of each specific GDC for a generic NLWR AR design as well as technology-specific criteria for sodium fast reactor and modular high-temperature gas-cooled reactor designs.

The current GDCs for plant licensing in the United States are based on LWR technology, that is, boiling and pressurized water reactors (BWRs and PWRs, respectively). Although 10 CFR 50 Appendix A provides some guidance in establishing PDCs for NLWR designs, it is the applicant's responsibility to develop the PDCs for its facility based on its design, using generic design criteria, NLWR design criteria, or other design criteria as the foundation. In Reg Guide 1.232 [12], the NRC describes NLWR design criteria to provide guidance for interpreting and developing PDCs for NLWRs. In that framework, applicants must consider matters related to public safety, include fundamental concepts such as DID, and identify and demonstrate how the reactor design satisfies necessary safety requirements. An important consideration in the development of the guidance provided in Reg Guide 1.232 is an acknowledgment of the benefits that can be obtained by risk informing the NLWR design criteria, to the extent practicable given the available design information and data.

Since 1946, the CNSC has been overseeing activities related to the application of commercial nuclear power in Canada. With growing national and global interest in new concepts for ARs (including SMRs), the CNSC has developed a strategy to tackle the challenges of regulating these technologies and prioritizing its regulatory efforts. The CNSC's regulatory readiness is built on three pillars: 1) a robust and adaptable regulatory framework, 2) use of risk-informed processes, and 3) access to a knowledgeable and capable workforce with adequate capacity and technical expertise [13].

Because the current regulatory framework in Canada is based on OE with Canadian deuterium-uranium (CANDU) heavy-water reactors, the rigorous application of related requirements to AR technologies presents a challenge. Therefore, the CNSC acknowledges the need to adapt its regulatory framework to align with the new technologies [13], applying a more objective-based regulation that provides flexibility to support licensing of non-water coolant / non-CANDU based reactor technology. The CNSC's regulatory readiness strategy includes the application of a risk-informed approach to regulatory review and decisions and to assessing alternative approaches to meeting current requirements [13].

Outside North America, several IAEA reports aim at providing guidance in developing a new technology-neutral safety approach and methodologies to define safety requirements for innovative reactor designs. These guidance documents recommend that new main pillars for the design and licensing of innovative nuclear reactors be developed following a top-down approach to reflect a newer, risk-informed, less prescriptive, technology-neutral framework. Although the overall strategy to define safety requirements to ensure a safe design for ARs is intended to be technology-neutral and technology-specific, providing a minimum set of safety requirements that must be achieved remain the pillars for licensing of any reactor design.

In particular, the current IAEA safety approach is based on four main pillars that are discussed in IAEA Safety Standard No. SSR-2/1 [14]. These are:

- Qualitative safety objectives: general nuclear safety, radiation safety, and technical safety
- Fundamental safety functions: confinement of radioactive material, control of reactivity, and heat removal from the core
- Defense in depth: several levels of protection, such as multiple barriers to the release of radioactive materials and provision of safety systems designed to ensure the safe shutdown of the reactor
- Probabilistic safety assessment (PSA[3]): application of PSA techniques along with deterministic methods

The IAEA has published a series of safety standards reports that can be used by regulatory bodies, government agencies, and organizations that design and operate NPPs. The requirements and recommendations included in these IAEA Safety Standards present current international practices and experiences for designing and licensing NPPs. Three levels of guidance are provided:

- Safety fundamentals: basic objectives and principles of radiation protection and nuclear safety
- Safety requirements: requirements that must be met to ensure safety
- Safety guides: recommended actions, conditions, or procedures for meeting safety requirements

In August 2003, the IAEA conducted a case study published as TECDOC-1366, *Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors* [15]. This study concludes that the existing safety approach's primary concepts could be suitable for new plants if correctly interpreted and formulated. The study suggests that implementing the DID concept in a methodical and comprehensive manner, along with "quantified" safety goals, could provide the confidence that an NPP design is safe and reliable and provides an adequate level of public safety. Based on this study, a new safety approach and a methodology to generate safety requirements for licensing of ARs are suggested. The approach is technology-neutral and less prescriptive, and it uses more risk-informed methodologies (see Section 2.2).

[3] In this report, the terms PRA and PSA are considered to be identical and are used interchangeably.

A general methodology for developing technology-neutral safety requirements for innovative reactor designs was developed by IAEA and published in 2007 as TECDOC-1570, *Proposal for a Technology-Neutral Safety Approach for New Reactor Designs* [16]. Experience highlights the importance of establishing safety goals, employment of DID concepts, and the benefits of integrating risk insights early in the iterative design process. Such an integrated approach will include the probabilistic elements of DID. This will help define the cumulative provisions to compensate for uncertainty and incomplete knowledge of accident initiation and progression, which is particularly important for AR designs, given the limited experience with these reactors. More recently, in 2022, an application of the IAEA methodology was assessed and published in TECDOC-2010, *Approach and Methodology for the Development of Regulatory Safety Requirements for the Design of Advanced Nuclear Power Reactors, Case Study on Small Modular Reactors* [17].

2.2 Role of PRA in Advanced Reactor Licensing

Advanced nuclear reactors, including SMRs, have unique designs and features that may require specialized PRA methodologies. In the United States, the NRC has developed specific guidance for PRA in advanced nuclear reactor licensing that includes the use of risk-informed, performance-based approaches to evaluate the reactor's safety performance.

The use of PRA in advanced nuclear reactor licensing can provide several benefits. It can help identify potential safety issues early in the design process, allowing for design improvements before construction begins; inform the selection of safety measures and help optimize their effectiveness; and demonstrate compliance with regulatory safety standards, thus providing broad confidence to stakeholders in the safety of the reactor.

NRC Reg Guide 1.233 [9] provides guidance to applicants who are seeking licenses to operate advanced NLWR NPPs. The safe design of NLWRs relies on fundamental aspects such as the selection of LBEs, the classification and special treatment of SSCs, and the assessment of DID. The guide sets forth a recommended approach for using a risk-informed and performance-based methodology in support of the licensing process. It endorses an industry-developed process and guidance provided in NEI 18-04 [8] as an acceptable approach for NLWR designers to use in their licensing applications. While the approach offers a standard method for LBE selection, SSC classification, and DID assessment, determining the suitability of specific technical requirements to meet NRC regulations or identifying new technical requirements from safety evaluations is done on a case-by-case basis. Like all NRC Reg Guides, this one provides an approach that the NRC has determined acceptable for complying with regulatory requirements; however, Reg Guides "are not substitutes for regulations and compliance with them is not required" [9].

The NEI 18-04 report [8] provides guidance to NLWR designers and applicants for developing the licensing basis for their designs. The NEI guidance uses a technology-inclusive methodology to identify LBEs, classify plant SSCs, and assess DID. This methodology provides a general approach for identifying the scope and depth of information that applications for licenses, certifications, and approvals should provide. In particular, NEI 18-04 emphasizes the importance of PRA and risk-informed decision making in the licensing process. It provides guidance on how to use PRA to inform LBE selection, SSC classification, evaluation of the effectiveness of DID, and determination of the need for additional programmatic controls. The core of the approach is the use of a set of frequency-consequence (F-C) criteria as measures to achieve different safety targets. Ultimately, the report provides guidance on how to identify performance-based PRA safety functions (i.e., functions responsible for the prevention and mitigation of an unplanned radiological release from any source within the plant) and to evaluate the effectiveness of SSCs and programmatic controls.

The methodology described in NEI 18-04 and endorsed in Reg Guide 1.233 offers a general approach for determining the appropriate scope and depth of information that applications for licenses, certifications, and approvals should provide for NLWR technologies. Due to the variety of NLWR technologies using different coolants, fuel forms, and safety system designs, defining a methodology is considered to be more appropriate and flexible than prescribing the specific content of the application. The methodology provides a structured approach to identifying the safety and risk significance of SSCs and associated programmatic controls that will be employed at the plant. The method focuses on identification of specific risks that are applicable to the specific NLWR technology and plant design as well as measures to address them. This approach is intended to promote more effective and efficient regulatory reviews.

Another example of a documented regulatory framework for AR deployment, the CNSC framework for risk-informed decision making, is described in REGDOC-3.5.3 [18]. The approach is based on risk-informed decisions or recommendations pertaining to licensing, certification, compliance, and the development of regulatory requirements and guidance. The CNSC allows applicants/licensees to apply a graded approach or to propose alternative methods to meet regulatory requirements and guidance. The key principles applied when using the risk-informed approach are that 1) regulatory requirements are met, and 2) sufficient safety margins are maintained.

In 2019, the CNSC and the U.S. NRC signed a memorandum of cooperation (MOC) to increase collaboration on technical reviews of AR and SMR technologies [19]. Under this MOC, a work plan was approved for exploring and seeking convergence on the regulatory approaches and guidance for applicants and regulatory reviewers in both countries. As an outcome of this cooperation, a report was produced documenting areas of commonalities and differences between the Canadian and U.S. approaches to safety assessments and licensing for ARs [20]. The work plan primarily concentrated on technical matters and the collaborative ability to carry out joint technical assessments. The report focused on safety analysis expectations and examined the technology-inclusive, risk-informed, and performance-based (TI-RI-PB) process developed as part of the Licensing Modernization Project (LMP) led by the U.S. nuclear industry and described in NEI 18-04 [8]. The joint CNSC-U.S. NRC report concluded that there is common ground in safety case assessment reviews and acceptance criteria, and that this can be used as a foundation for one regulator to leverage the technical reviews of the other. Legal and policy matters related to each country's regulatory framework were not included in the scope of the report and would require individual consideration by the respective regulators when making independent regulatory judgments and determinations. The ultimate goal of the collaboration is only to achieve joint technical reviews.

In Europe, IAEA TECDOC-1570 [16] proposes a new safety approach for new NPPs based on the review of existing pillars to include consideration of new technologies and incorporation of probabilistic considerations. In particular, the safety goals are identified in terms of consequences as a function of likelihood of occurrence. The goals are derived from the overall safety objectives ("to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defenses against radiological hazards") and are evaluated in quantitative terms by means of an F-C diagram. Events with frequencies and consequences above a given value that occur during the normal or abnormal operation of a nuclear plant are not accepted. This overall approach is independent of the new technology, and specific assessment based on the reactor type must be done to identify specific safety aspects related to that reactor design. Therefore, in this context, PRA plays a key role in identifying aspects of the design of the plant that could lead to the occurrence of such events. One of the objectives of the approach is "to develop a 'safety-driven' plant design which complies with the new Safety Approach and the newly derived safety requirements." Consequently, specific consideration should be given to the practicability of design improvements as part of implementation and execution of the process.

In terms of safety functions (i.e., reactivity control, heat removal, and confinement of radioactive release), the DID approach is a fundamental pillar that provides assurance that the functions will be met with a high degree of confidence. Overall, the concept of DID integrates considerations both deterministic (such as number of defense levels, diversity in providing the necessary functions, and their independence) and probabilistic (such as equipment reliability and probabilistic targets) to evaluate the adequacy of design provisions for each defense level and ensure their consistency of implementation. Additionally, to determine the safety classification of plant SSCs, a safety assessment model of the entire plant safety architecture should be used as a complementary method, without being influenced by any preconceptions of what may be important for safety. This model can also be used to assess the contribution of each provision to the overall safety of the plant and to identify any provisions that require strengthening. The model allows for a direct assessment of the value of improvements to these provisions.

The methodology presented in the IAEA reports facilitates an evaluation of overall plant performance, utilizing both deterministic and probabilistic approaches. The deterministic approach outlines the necessary physical performance requirements, while the PRA model ranks component reliabilities and scenarios by likelihood of occurrence. As a result of this process, the most likely events and their characteristics in terms of physical performance, reliability, availability, and independence are identified. Technical design specifications are generated from physical performance considerations, and quality requirements are generated from the assumed reliability.

Risk and risk-informed processes are considered in this framework. These processes rely on a PRA and the definition of probabilistic success criteria. To assess any reliability target across the full range of operating and accident plant states, PRA methodologies will need to be modified and extended. As developed and applied for LWRs, a Level 1 PRA is required for a single core damage target, while a Level 2 PRA is needed for a release of a radiological source term extending beyond the installation. Due to the wide diversity of AR designs, standard concepts and assumptions that are accepted for use in an LWR PRA may need to be modified to address the unique aspects of these designs. For example, for reactors that utilize liquid fuel cores, the concept of core damage may not be meaningful in the assessment of plant risk. Another example is the current assumption of a 24-hour mission time for PRA analysis used for LWRs. Note that the potential for such changes would not necessarily be limited to NLWR ARs; LWR SMR designs also incorporate features, such as reliance on passive systems, that could require changes in the PRA for these designs. Moreover, uncertainties should be considered in the PRA modeling, especially for innovative designs where reliability data may be limited. The IAEA advised use of conservative approaches in such cases [16, 17]. Other uncertainties to consider include modeling methods, human reliability performance, and data uncertainties.

TECDOC-2010 [17] presents an application of the methodology discussed in TECDOC-1570 [16] that considers an integrated risk-informed, objective-oriented, performance-based assessment approach that can be used for the development of regulatory safety requirements for the design of advanced NPPs using different technologies, including SMR designs. The methodology describes how the safety requirements established in SSR-2/1 (Rev. 1) [14] for NPP design can be adapted to be applicable to advanced NPP designs, with a particular focus on application to SMRs. The design safety features for reactor technologies used in the most typical SMRs are covered in a technology-neutral manner (for both water-cooled and non-water-cooled SMRs). The TECDOC-2010 document provides examples illustrating the adaptation of design safety requirements from SSR-2/1 (Rev. 1).

In particular, the current safety requirements for NPP design, as outlined in SSR-2/1 (Rev. 1) [14], were developed based on the experience with large water-based NPPs (BWR, PWR, and CANDU reactors). Therefore, the design of SMRs may present challenges in terms of complying with existing safety requirements. Additionally, the design safety requirements for water-cooled reactors may not be sufficient for advanced NPP designs incorporating other technologies, including SMR designs, and may require adaptation to be technology-neutral, inclusive, and specific, as proposed in Idaho National Laboratory report INL/EXT-14-31179 [21] and described in NRC Reg Guide 1.232 [12]. The application of design safety requirements often needs to be graded to accommodate various technological and regulatory considerations, using an integrated risk-informed, objective-oriented, performance-based approach. This approach aims to ensure the highest level of safety that is reasonably achievable in a structured manner by covering all areas impacting nuclear installation safety, such as probabilistic and deterministic factors, human and organizational aspects, and the interface between safety and security. The use of an integrated risk-informed decision-making (IRIDM) approach for evaluating the appropriateness of developing, updating, or adapting regulatory design safety requirements has the advantage of being both technology-neutral and applicable to all regulatory approaches, including prescriptive and objective-oriented approaches. Details about information on the application and factors to be considered in the IRIDM process, with examples, can be found in IAEA TECDOC-1909 [22]. The integrated risk-informed, objective-oriented, performance-based approaches together have the advantage of ensuring a comprehensive and structured approach to developing, adapting, or updating regulatory design safety requirements [17].

The proposed regulatory guidance and frameworks from the U.S. NRC, CNSC, and IAEA apply PRA as a key element in the overall technology-inclusive and risk-informed approaches. In the guidance endorsed in the United States, PRA techniques are used to risk-inform the design by establishing frequencies and consequences of LBEs and to classify SSCs that prevent or mitigate events that potentially can impact public safety or ensure adequate margins to the F-C target curve used for plant licensing.

The approach taken by the CNSC also involves using PRA techniques as an important input for identifying events and classifying systems. Probabilistic inputs are used in conjunction with other deterministic safety analyses and requirements to achieve a comprehensive understanding of the safety of the system. To this end, the CNSC requires Level 1 and Level 2 PRAs, which estimate the frequency of core damage and the frequency of both large and small releases that could result in temporary evacuation or long-term relocation of the public located in the vicinity of the plant site. The CNSC has established safety goals specifying that the sum of frequencies for any sequences with releases exceeding the safety goal limit should not exceed 10^{-5} per year for small releases and 10^{-6} per year for large releases.

The IAEA also uses PRA techniques in its proposed methodology to license advanced NPPs. In particular, safety objectives are evaluated using an F-C diagram in a manner similar to, though less specific than, that described in NRC-endorsed industry-developed guidance (i.e., NEI 18-04 [8]). In this risk-informed F-C framework, events with frequencies and consequences above a given set of thresholds would not necessarily be considered unacceptable from a regulatory/licensing standpoint. The DID framework also relies on risk and risk-informed processes that are combined with deterministic considerations for the identification of appropriate defense levels. Ultimately, the methodology combines deterministic (necessary physical performance requirements) and probabilistic aspects (PRA model used to rank component reliabilities and scenarios by likelihood of occurrence) in a technology-neutral, less prescriptive fashion.

The safety case demonstration specified by the IAEA for ARs that will be deployed in different countries is intended to result in use of a similar framework to identify events, classify them, and ensure that potential consequences of plant accidents meet a consistent set of regulatory standards. The IAEA also intends that all countries use equipment classification and DID approaches based on objective principles. It is recognized that there will exist some differences in implementation in different countries, but these would be related to country-specific implementation rather than safety policy or philosophical approaches. One of the general observations from this review and comparison indicates that there is significant common ground in safety case assessment reviews and acceptance criteria. This common ground can be used to support use of technical reviews by one regulator to inform the independent regulatory findings and decisions of other regulators, thus allowing for more efficient and consistent regulatory reviews to support licensing and deployment of ARs in different countries.

2.3 Literature Review of Development and Use of Advanced Reactor PRA

In this section, an overview of the main literature relevant for the use of PRA in ARs is provided. The review focuses on different aspects relevant for the development of AR PRA, such as:

- Use of PRA in early stages of reactor design
- Identification of potential initiating events
- Support of risk-informed design
- Performance of human reliability analysis (HRA)
- Assessment of the reliability of passive safety systems
- Data and OE
- Multi-unit risk
- Digital I&C
- Emergency preparedness requirements
- Risk-informed licensing and regulatory framework

The use of PRA in the early stages of AR design is a key topic among both researchers and developers of ARs. Level 1 PRAs or both Level 1 and Level 2 PRAs may be requested by the applicable regulatory authority for each new reactor project. Level 3 PRAs are typically not developed or requested by the regulatory authorities for new reactors. However, a Level 3 PRA could support the definition of exclusion zones [25].

Furthermore, in the United States, the LMP has developed a methodology that uses standard safety analyses like Process Hazards Analysis (PHA) and PRA to support various safety applications during the design and development of ARs, including evaluation of design alternatives, selection of LBEs, classification of SSCs, and assessment of the adequacy of DID [8].

In this regard, the general intent is to enhance safety by adopting a safety assessment approach that facilitates up-front incorporation of safety into the system design, rather than just adding safety measures to compensate for limitations [23]. For example, in Chisholm et al. [24], a methodology was proposed to identify hazards in AR designs in a flexible and comprehensive way. The methodology was technology-neutral and used to construct a PRA model for the Molten Salt Reactor Experiment (MSRE) as a demonstration. A Hazard and Operability Analysis (HAZOP) and Failure Modes and Effects Analysis were performed on parts of the MSRE as a test case to inform the construction of event trees and fault trees for most of the facility's mechanical systems.

An early integration of safety assessment into the design process indicated the process can be enhanced to support early, effective engagement with regulatory authorities [26]. In a 2020 paper [27], the authors focused on identifying all potential initiating events (PIEs) by considering all possible scenarios that could lead to failure in a system design. These papers indicated that, while there are generic lists of PIEs available for LWRs, there is not enough OE with liquid fluoride molten salt reactors to develop a similar list. However, the authors concluded that, using a PRA-based methodology, risk-significant PIEs can be identified during the analyses of the MSRE design, and changes to the design itself can be made.

In 2019, EPRI and Vanderbilt University initiated a project that aimed at consolidating existing tools, methods, and best practices into a practical Safety-in-Design (SiD) approach for ARs [28]. This approach is intended to support a design-to-licensing process that is risk-informed and performance-based. In particular, the approach is based on the use of qualitative and semi-quantitative PHA methods to provide a viable starting point for developing the fundamental elements needed for quantitative design assessments, such as PRAs. The goal was to leverage risk-based insights early in the design process and gradually develop and quantify the safety design basis as the reactor design evolved.

The researchers concluded that the early use of the SiD method is effective in identifying hazards for NLWR technologies that are currently under development. The use of PHA was found to provide a more comprehensive and systematic approach to accident scenario development than historical research. The SiD method can be applied to meet a broad range of needs, from reactor conceptual design to development of a non-radiological test apparatus in preconceptual design. The early application of SiD principles promotes early graded implementation of programmatic practices that will need to be fully developed and in place to support the future larger organization when the design reaches high maturity. However, the risk metrics should be expected to evolve over time; those used to investigate qualitative risk at an early stage of design may be different from those used to investigate the quantitative risk of a more mature design.

In addition, a 2020 EPRI report [29] documents a series of safety assessments using the SiD methodology on subsystems and design features of the MSRE. The report describes a process for integrating safety into the design of advanced nuclear reactors. Key findings include the effectiveness of the SiD methodology in identifying hazards for NLWR technologies and the need for standard hazard identification tools to be improved to identify slowly developing threats (e.g., for the MSRE, corrosiveness of the fuel salt, which may result in a fuel leak or off-gas leak through a pressure boundary). The report also highlights the need for thorough hazard analyses to identify new risk-significant sequences and/or information to support design decisions.

Overall, the early application of safety assessment tools and methods during reactor system design can provide value and facilitate design maturation by identifying important knowledge and design performance gaps, allowing changes to be incorporated with the least impact on cost, schedule, and licensing. The knowledge and insights gained during this application led to enhancements in the approach and plans for advancing the method to conduct a comprehensive and successful SiD evaluation of advanced nuclear reactor designs [28].

Another aspect to be considered during the early design of an AR is the assessment of operator actions and their impact on safety. In the United States, a complete assessment of risk, including HRA, is required as part of the final safety analysis report for any licensing application using any of the potential licensing pathways (i.e., 10 CFR Parts 50, 52, or 53). In any nuclear application, the human factors engineering program will be an essential component of the licensing application, ensuring high personnel performance and providing important information and data used to support the development of the HRA.

Performing HRA during later design stages can capture human failure events before, during, and after the initiating event; however, as discussed by Hamza and Diaconeasa [30], it falls short of providing insights into the design itself in an iterative design life cycle. Therefore, Hamza and Diaconeasa introduce a framework for incorporating HRA early, at either the preconceptual or the conceptual design phase. The proposed framework outlines a process for distinguishing critical operator actions that can contribute to safety from operator actions that do not affect risk (and therefore do not need to be modeled in the HRA). The framework's results are used to inform the design of safety-significant operator actions, which then are further used to update the design when appropriate. Subsequently, the framework can be reapplied to evaluate the effect of the design changes on human reliability, using information from the updated design. Hamza and Diaconeasa [30] apply the framework to X-energy's Xe-100 preconceptual design, highlighting key sequences that require further investigation and potential design/procedure updates.

The PRA Standard for Advanced Non-LWR NPPs [6] specifies technical requirements for the 18 elements needed to create a full-scope PRA, with two PRA capability categories, CC-I and CC-II, based on plant-, site-, or design-specific models. Human actions are incorporated into all 18 PRA elements. One specific element is responsible for HRA, which should be included in both capability categories to ensure the identification of risk-significant events.

The reliability of passive systems is another area of research that will be fundamental for the development of PRA for ARs. Over the past decade, a substantial literature on the assessment of passive system reliability has been developed; EPRI reports [31, 32] provide a useful summary of the initial research. More recent studies [33, 34, 35, 36] discuss different methodologies to assess the reliability of passive safety systems, on which many advanced nuclear reactor designs rely. Such systems depend on natural driving forces, such as natural circulation, gravity, or internal stored energy, rather than on external power sources, to prevent and mitigate accidents [36]. To ensure the effective and stable operation of these passive systems, it is important to evaluate their reliability using reliable methods [33]. This is done by assessing the physical margins (such as thermal-hydraulic [T-H] or reactivity margins) within which the passive system can operate.

The assessment methods here are different from those typically used in traditional PRA because the driving forces of passive systems (which are limited by natural forces such as gravity) are much smaller than those of the active (e.g., forced cooling) systems used in LWRs. Operational phenomena and conditions (such as fouling of heat transfer surfaces, corrosion, or aging effects) can have a much greater impact on passive than on active system performance and capabilities. Some of the affecting factors, such as heat transfer coefficients and pressure losses, can be subject to considerable uncertainties. These uncertainties can impact the T-H performance of the passive system, which in turn affects its reliability.

Thus, the characterization of the physical margins can provide essential information for both licensee and regulatory authorities to support decision making for the licensing and operation of advanced NPPs [34]. Ultimately, the characterization of these margins and assessment of what parameters influence the performance of passive systems are essential in the physics-based reliability analysis [33]. Nevertheless, it can be more difficult to assess the reliability and performance of passive than of active safety systems due to a lack of experimental or operational data on phenomena that rely strictly on natural forces [35].

To address these uncertainties, different methodologies, such as the reliability evaluation of passive safety systems (REPAS) [37], reliability methods for passive safety functions (RMPS) [38], and analysis of passive systems reliability [39], have been developed. These methodologies involve several steps, including identifying and quantifying sources of uncertainties, propagating these uncertainties through T-H models, and incorporating passive system unreliability into accident sequence analysis. The referenced methodologies have some features in common, the primary attribute being adoption of a load-capacity model (prevalent in the structural engineering field) for characterizing likelihood of system success. However, they differ on certain issues, such as treatment of model uncertainties and deviation of geometric and process parameters from their nominal values [40].

The primary issue impacting passive system reliability is that, due to the smaller margins between the designed operational parameters and conditions necessary for successful performance to meet the intended safety function, deviations of process parameters from their nominal values can greatly affect system performance. Such deviations can increase or decrease the probabilities of occurrence of extreme events, as well as the failure rate of components. Moreover, passive system components can fail in intermediate conditions during operation, contrary to the traditional assumption of binary state failure. Current methodologies may not adequately address the dynamic failure characteristics of these components. Additionally, the reliability analysis of passive systems should consider the dynamic variations of independent process parameters, such as atmospheric temperature [40].

The performance and reliability analysis of circulation-based passive systems based on best estimate codes using phenomenological simulations must therefore incorporate the significant uncertainties in an appropriate manner. Given these uncertainties, a large number of calculations with best estimate T-H codes likely will be needed to assess system reliability. In summary, the primary challenge related to evaluation of passive SSCs is the characterization of SSC reliability and the impact on the overall risk profile in a manner that provides confidence in the results to support risk-informed decision making.

The impact of external events (e.g., seismic events, external flooding, high wind, issues at nearby facilities) on passive safety systems needs to be taken into consideration in the overall evaluation of their reliability. The hazards caused by external events are considered in the safety and design of NPPs. For example, NPPs are normally designed to resist specified levels of earthquake or flooding events. One important factor in assessing sites for the deployment of new reactors is the identification and characterization of external events used as design basis events, including events that might preclude the use of the site for its intended purpose [41].

External events have the potential to change assumed system boundary conditions. For passive systems, boundary conditions may be important to induce the driving force, and changes of these conditions could lead to failure of the system to perform the required function [8]. Therefore, generally speaking, it is recognized that external events need to be included in the design of ARs (for example, see IAEA TECDOC-1487 [42]). Because safety requirements for design in relation to external events may not be commensurate with the risk involved, some external events may be excluded from the design through deterministic and probabilistic screening. One of the main conclusions of TECDOC-1487 is that "the development of an external event PRA in parallel with the early plant design may help to identify the vulnerabilities as well as potentially overly conservative design features at an early stage, leading to a well-balanced and cost-effective improvement in safety." To define safety goals and licensing criteria, PRA can be used as a tool to characterize the risk due to external events and demonstrate that the safety goals provide assurance that the annual health effect and dose limits are achieved.

Hence, an issue that needs to be clearly addressed moving forward is the balance between the cost and benefit of developing detailed external events PRA models with insights that can be used to adequately and robustly screen external events that may otherwise be expected to be fully included. The balancing process may consider, for example, whether such an approach is feasible and justifiable, and whether a graded approach can be provided that accounts for increased margin to safety goals with sufficient consideration of uncertainty and degree of confidence.

Another important aspect to be considered is related to multi-unit risk. The Fukushima Daiichi incident showed the possibility of accidents involving multiple reactor units and spent fuel pools experiencing nearly concurrent core damage. This accident's progression was affected by complex interactions among the facilities and operator actions taken to protect each one. As a result, there is a need to evaluate site risk in an integrated way that considers the potential for concurrent accidents involving multiple installations [43, 44, 45]. This requires the integration of risk contributions from different sources, hazard groups, and plant operating states, which may involve qualitative and quantitative information. Whole-site PRA is a supporting tool and subset of whole-site risk assessment that complements other factors in risk management [43, 46]. A major effort was developed by EPRI on the topic of multi-unit risk [47]. While the resulting report focused on existing large commercial operating nuclear reactors, the general approach and specific topics (e.g., external events) will relate more broadly to nuclear technology that relies on multiple units (or modules) on site. While the specifics will be addressed in future work, the critical issue is: to what extent are multi-unit/multi-module risk assessments needed, practical, and/or useful for risk-informed applications of nuclear technology? Further, if they are needed, how can the scope and interpretation of their results be clearly integrated for purposes of risk-informed decision making? The EPRI report covers risk-informed decision making and technical issues and challenges applicable to multi-unit or multi-module facilities, such as initiating events selection, accident sequence modeling, quantification of accident sequences and site-based risk metrics, characterization of accident progression, release metrics, risk integration, and interpretation.

The use of computer systems has been essential to the design, construction, and operation of nuclear reactors and will continue to be so in the future [48]. The development of advanced nuclear reactors, from initial concepts to detailed design, licensing, and operation, is heavily reliant on computational models. To enhance the design process, an integrated framework for reactor modeling is necessary to enable smooth communication, coupling, automation, and continuous development. By linking key performance metrics (such as optimal fuel management, peak cladding temperature during design basis accidents, and levelized cost of electricity) to design inputs, an exceptional level of design consistency can be achieved. Additionally, with the aid of high-performance computing, thousands of integrated cases can be simultaneously analyzed, enabling complete sensitivity studies and efficient evaluation of various design tradeoffs for the full system [48].

Software models have progressed from simple to highly sophisticated and are now being employed as integrated full-system tools. As technology advances, artificial intelligence will likely be integrated with these models to enable the automation of physically consistent conceptual designs based on user-input characteristics and performance targets. It is conceivable that such models will be linked to I&C systems in operating plants to ensure efficient, safe, and seamless operations. An example of these tools is the Advanced Reactor Modeling Interface code system deployed to support the TerraPower Traveling Wave Reactor design [48].

EPRI has developed methods to assess performance and risk impacts related to digital I&C systems. This issue is significant due to the use of digital technology to perform essentially all I&C functions for advanced plants, and there have been substantial challenges related to obtaining efficient regulatory reviews and approvals when digital upgrades have been deployed to replace the original analog systems that interface with plant safety systems in the current fleet. To address this issue, EPRI has developed the Hazards and Consequences Analysis for Digital Systems (HAZCADS) approach [49].

Emergency preparedness requirements are also a topic to be considered for the licensing and deployment of AR technology. Emergency planning zones (EPZs) can be defined according to the spatial distribution of postulated potential radionuclide releases. Although there may be design differences between countries, an EPZ is generally defined as the area(s) around a nuclear site where arrangements are made to protect the public and the environment during the occurrence of a nuclear emergency [50].

ARs have different design aspects than current reactors (in some cases, significantly reduced nuclear material inventory), and these call for emergency planning that differs somewhat from that currently implemented in the industry. In general, ARs have been designed to reduce or delay the release of radioactive material during postulated accidents. These features are intended to reduce risk to the public and to potentially reduce or eliminate some of the emergency plan and evacuation requirements [51]. SMRs have unique features that make the overall plant safer, with radioactive release risk reduced in both likelihood and consequences. The trend in expectations is that some ARs may not need relocation or evacuation measures similar to those required for existing large commercial NPPs (see, for example, the conclusions in an IAEA project [52]). Hence, there is an ongoing evolution that may allow scaling emergency-related site requirements while maintaining safety for the public equal to or better than that offered by the existing fleet of reactors. This scaling could positively impact the nuclear industry due to several additional benefits, such as the following:

- Increased number of possible sites for reactor deployment, even in areas with higher population density than typical for the current reactor fleet
- Reduced need for additional major infrastructure such as grid and cable connections because the energy production site can be located in the vicinity of energy usage, ultimately reducing cost
- Ability to use the NPP not only for electricity production but also for additional applications such as co-generation, district heating, or desalination
- Alignment of nuclear power emergency planning with expectations similar to those for commercial industry facilities

In the United States, the NRC is currently working on updating the regulatory framework to accommodate the licensing of NLWRs. One approach that researched a potential licensing strategy for NLWRs was presented to the U.S. Congress in 2008. It included a technology-neutral, risk-informed, and performance-based (TN-RI-PB) framework that was to be applied to a high-temperature gas-cooled reactor as a demonstration. The framework was based on a series of white papers addressing safety issues such as event selection, safety classification of SSCs, evaluation approaches for DID, and use of PRA in the decision process [52]. In this area, Idaho National Laboratory (INL) performed initial regulatory analyses to support NRC efforts to prepare for licensing of ARs. These analyses investigated:

- "Right-sizing" of emergency planning requirements for ARs
- Specification of a license structure for power plant sites with multiple reactor "modules"
- Development of a mechanistic approach for defining and establishing the radiological source term
- Description of multibarrier radionuclide containment approaches

The INL regulatory affairs team also managed the development of proposed guidance for advanced NLWR design criteria development in support of a joint Department of Energy-NRC initiative. The technical bases contained in the Next Generation Nuclear Plant TN-RI-PB white papers provided the foundation for the LMP's proposed licensing framework [52].

Although this literature search endeavored to be comprehensive, the AR stakeholder community identified a number of issues (see Sections 2.1. and 2.2) for which research or guidance was not found in the literature surveyed. In particular, the following issues do not appear to have been investigated in any systematic or substantive manner for ARs:

- Methods to assess the frequency of occurrence of release of large amounts of radioactive materials (Level 2 PRA) resulting in a large release frequency (LRF)/large early release frequency (LERF) and consequent impacts on the public (Level 3 PRA)

- Methods to evaluate risks not related to nuclear safety

It is assumed that the methods developed and applied to analyze these issues for the current fleet of LWRs will be applied for ARs. However, there exist significant uncertainties related to the degree to which existing methods will be complete and efficient for ARs.

One critical element related to ARs is the requirement that these reactors provide safe and economically competitive energy throughout the life cycle of the facility. From the perspective of the AR community, particularly reactor vendors and plant operators, management of enterprise risk is needed to ensure profitable economics, particularly through the licensing, construction, commissioning, and operational phases, of ARs with different dynamics than large projects such as those in the current commercial nuclear reactor fleet. Achieving on-time, on-budget construction with successful high-capacity operation of the plant after commissioning, especially for the initial plant deployments, is considered to be essential to achieve widespread acceptance and adoption of ARs. Recently, EPRI has initiated general research in enterprise risk management (ERM) to help address this issue and has reported its initial results [53].

3 RESEARCH APPROACH, ANALYSIS, AND ROADMAP

The objective of this research is to identify and prioritize technical issues related to the conduct of risk assessments of advanced nuclear plants. Given the state of development and plans for licensing/deployment of ARs, greater consideration and priority are given to identifying and conducting research to address gaps that support near-term licensing actions for ARs. This chapter describes the approach used to identify and prioritize areas where research is needed to address issues related to the development, execution, and evaluation of PRA applied to ARs. This approach is described in Section 3.1. Results from the assessment, along with characterization and prioritization of the identified issues, are described in Section 3.2. From these assessments, a research roadmap to address the identified issues is proposed in Section 3.3.

3.1 Research Approach

As described in Section 2, significant work has been performed by both the nuclear power industry (i.e., AR designers and potential plant owner/operators) and responsible regulatory agencies to specify requirements for licensing ARs. In recognition of the wide variety of reactor designs that are being developed, these requirements and associated implementation guidance are structured to be technology-inclusive and risk-informed. This guidance serves as a useful starting point to identify potential areas where research into data, methods, and tools may be beneficial (and even necessary) for the effective and timely risk assessments needed to support the licensing and deployment of ARs.

Issues related to the definition and execution of PRA and its application to the design, licensing, construction, and operation of ARs were identified and assessed through a series of ongoing interactions with stakeholders, including two distinct activities. First, EPRI conducted interviews with staff from organizations that are actively engaged in the design and possible deployment of ARs. These initial discussions included both AR developers and utilities (i.e., potential plant owner/operators), covered a broad range of technologies and design concepts, and included both domestic (U.S.) and international organizations. The purpose of the interviews was to permit each organization to identify issues related to risk assessment needs for ARs from its perspective. The interviews were conducted in an interactive format between EPRI staff and staff from the interviewed organizations who are involved in various aspects of AR design, operation, and risk analysis. A meeting agenda was used to guide the discussions, and the information provided was captured in notes taken by EPRI and provided to the host organizations for review. The notes from the interviews were then evaluated to identify recurring themes identified by multiple organizations.

The second activity was an EPRI stakeholder workshop on needs associated with performing and applying PRA for ARs. The workshop was hosted in EPRI's Charlotte, North Carolina, office. The primary objective was to exchange knowledge and opinions on risk analysis methods and tools that will be needed to support the development and deployment of ARs. The outcome was input to EPRI research planning to identify and address critical issues related to risk assessment and management for ARs.

In addition to these two formal information collection activities, other exchanges of information have been considered in the development of this report. These include:

- Participation in NEI and IAEA AR activities
- Public meetings and documents related to NRC AR licensing initiatives
- Ongoing one-on-one interactions with AR stakeholders (e.g., vendors, utilities, national laboratories)
- Participation in other industry meetings and conferences related to PRA and ARs

Key results and insights from the stakeholder interviews and workshop are discussed in the next section. The results were evaluated to identify issues that could be addressed/resolved by the application of EPRI research.

3.2 Collection and Analysis of Stakeholder Input

In Section 3.2.1, results from interviews conducted by EPRI with individual AR stakeholders are discussed. The results obtained from the workshop are discussed in Section 3.2.2.

3.2.1 Stakeholder Interviews

A universal concern expressed by both reactor vendors and potential plant owner/operators was the lack of OE and data for AR designs; this was a particularly critical concern for NLWR designs. The issue has also been raised by regulatory authorities. For example, it engendered significant discussion at a recent IAEA International Conference on Topical Issues in Nuclear Installation Safety (October 2022). In addition to this global issue, several specific technical issues were identified by multiple organizations that participated in these discussions. These include the following:

- Methods to evaluate uncertainties related to the occurrence of long-duration events. PRAs associated with ARs are anticipated to require analyses that relate to different end states and potentially longer durations (i.e., greater than the current 24-hour mission time typically assumed in LWR PRAs).
- Methods to perform HRA related to actions that may be required in response to long-duration events (e.g., actions several days after the onset of the event).
- Methods to address changes in the role of operators at ARs compared to current LWRs (e.g., more focus on monitoring and supervision and less need to perform procedural actions to respond to abnormal events, transients, or accidents). This topic is potentially significant because many AR designs may have no requirements for safety-related operator response actions. It is also potentially significant to address regulatory concerns related to the need to analyze errors of commission (EoCs) in HRA because these could become the dominant source of human error for some AR designs.

It is expected that some of the issues related to operational data will be the subject of experimental work and initial prototyping. However, significant uncertainties likely will persist for AR designs until several commercial reactors using the various designs have been built and operated for a period of time to build an OE base. Additionally, as specific plant designs become more complete and reactor vendors gain more feedback from interactions with operating utilities and regulatory authorities, it is likely that additional areas requiring research and development related to plant PRAs will be identified.

In the interviews conducted by EPRI, multiple potential plant owner/operators identified analysis of passive systems as a concern. In particular, analyses performed to support initial plant licensing have generally assumed clean systems (e.g., no corrosion or fouling) to demonstrate adequate system performance. However, from the perspective of plant operations over the projected plant lifetime, the capability of plant maintenance and aging management programs to effectively monitor and control the impact of failure mechanisms on passive SSCs requires attention in order to maintain the very low risk profiles expected from AR designs. Additionally, if PRA is relied upon as an integral component in operational decision making, methods to account for any degradations in operational performance of passive systems likely will be needed to ensure the PRA accurately reflects the “as-built, as-operated” plant, as required by the various PRA standards [3, 6] and regulatory guidance associated with risk-informed programs (such as NRC Reg Guide 1.200 [54]).

An additional issue identified during the EPRI interviews was the need for methods and guidance related to the evaluation of risks associated with the planned colocation of multiple units/modules at a single plant site. This issue was identified primarily by potential plant owner/operators; most of the reactor vendors did not raise it. It is noteworthy that the intended “standard” deployment consists of multiple modules for most AR designs. This topic also is relevant to the existing fleet of operating reactors. In this regard, EPRI recently developed an integrated framework for addressing multi-unit risk at NPP sites using a graded approach with the objective of supporting risk-informed decision making by including contributions from multi-unit risk [10].

Interviewed stakeholders also identified an issue specifically applicable to NLWRs. As indicated previously, decades of operation of LWR technologies have developed a consensus on the risk metrics to be used for these technologies, particularly the use of CDF and LERF as figures of merit. However, for NLWR AR technologies, the LWR metrics may not be meaningful or useful for either regulatory or operational/management purposes. The recently published NLWR PRA standard [6] uses frequencies of various releases and consequences of radioactive materials as the figures of merit. This supports the current use of an F-C approach, which has been developed and included in NEI 18-04 [8], to support the licensing of NLWR ARs. Although this F-C curve has demonstrable links to regulatory bases associated with current licensing requirements in the United States, it has not achieved a consensus among all stakeholders (in particular, regulatory authorities across the world) as a basis for NLWR AR licensing. In addition, methods are needed to translate the approach into processes and procedures that can be used in day-to-day plant operational risk management. Because the F-C relationship as currently constructed is multidimensional, its translation into operational programs is potentially more complex and may present challenges to address prior to commercial operation. These challenges cannot be met without addressing the gaps in methods, approaches, and data needed to perform a PRA that complies with the requirements of the standard and supports the level of rigor needed to use the F-C approach.

3.2.2 Advanced Reactor Risk Workshop

The primary objective of the workshop was to exchange knowledge and opinions on risk analysis methods and tools that will be needed to support the development and deployment of ARs. The results of the workshop provided necessary input to EPRI research planning. The following specific research planning outcomes were identified:

- Determination of the readiness of current PRA methods and tools for use in ARs and identification of technical gaps that could be resolved through EPRI research
- Development of a research roadmap to guide EPRI research to address identified gaps and ensure readiness of PRA methods and tools to be applied by the AR community
- Execution of the research roadmap to address the identified gaps in a time frame that supports proposed schedules for AR licensing, deployment, and operation

The workshop consisted of presentations by a variety of AR stakeholders, including AR vendors, utilities (representing potential plant owner/operators), engineering support organizations, and universities (representing the research community). EPRI staff from across the nuclear sector provided discussions on topics of interest to AR stakeholders, including EPRI risk software, reliability integrity management (RIM) programs, HRA, digital systems, and ERM. In addition to the presentations, there were specific sessions for discussions among the workshop participants to exchange experiences, including identification of issues/challenges requiring research to resolve or support more efficient/effective approaches to address them. A summary of the key points made during the stakeholder presentations and the results of the discussion sessions is provided below.

- Joint presentation from AR vendor #1 and U.S.-based nuclear utility:
  - Availability of component reliability data is limited (e.g., liquid Na-cooled reactor component reliability data used for reliability assessments were derived from use of salt-wetted components as a surrogate).
  - Traditional static methods do not lend themselves well to analyzing some conditions applicable to NLWR designs (e.g., fuel salt/off-gas leaks).
  - The definition of a “safe stable end state” can be challenging given the long duration of postulated events (e.g., there may exist event end states where the reactor remains critical).
  - Development of the capability to automate population of the F-C curve used for safety/risk assessment and licensing submittals would be beneficial.

- AR vendor #2:
  - Early involvement with PRA at the design stage provides the opportunity to identify and address tradeoffs early, thus improving the plant design and effectively controlling costs.
  - In applying the guidance provided in NEI 18-04, definition of an additional analysis class would be beneficial to support analysis and decision making. As a result, an "incredible event" classification, defined as comprising events with a mean frequency less than 5E-7/yr, was developed. This classification is used to track these events for possible inclusion as beyond design basis events based on assessments of uncertainty.
- AR vendor #3:
  - Challenges include the conduct of design peer reviews and the use of relative risk measures to support decision making.
  - Challenges also include the methodology used to assess reliability of passive systems.
  - This AR vendor developed a multi-module risk assessment approach that applies adjustment factors. It noted that there is no clear distinction between multi-unit and multi-module designs; thus, development and specification of requirements applicable to multi-module designs may be problematic.
- University participant: This participant conducted EPRI-sponsored research to support the demonstration, maturation, and adoption of processes to support AR deployments. The key approaches considered included the following:
  - Safety-in-Design (SiD)
  - Systems Theoretic Accident Model and Process (STAMP)
  - Simple Improved Repository Risk Assessment Measure (SIRRAM)
  - Model-Based Systems Engineering (MBSE)

  Application of SiD was demonstrated in case studies performed on portions of the design process for the MSRE [29].
- Risk-informed engineering services provider:
  - ARs will be expected to achieve production and economic performance that are economically competitive with those of other sources of electricity production and that meet established requirements for reliability and availability. In particular, advanced plants will need to operate with very high levels of safety and economic performance from the time of initial commercial deployment (i.e., for widespread adoption of ARs, there will likely not be tolerance for a substantial learning curve to achieve the required performance objectives). This condition will require a much broader, more comprehensive, and more integrated approach to risk management than currently is employed for operating reactors. As an example of the need to integrate both safety and economic risk management, some AR designs have unique provisions to address accidents that would prevent core damage and any radioactive impacts to the public, yet they also would result in the likely loss of the plant as an economic asset.

- Published results of the evaluation of risk for ARs consistently show very low values for the metrics used for internal events at LWRs (e.g., CDF, LRF/LERF). As a result, external hazards will likely provide a substantially greater fraction of the risk to the public than they have for the current fleet of LWRs. The current state of knowledge is less developed for potentially dominant very low frequency external hazards (phenomenology, event occurrence frequencies) than it is for internal events. Therefore, the fractional contribution of external events to overall risk not only is larger but also will have greater uncertainty (in relation to the values of the risk metrics) than it does for the existing fleet of operating plants, which have a more mature understanding of their dominant risk contributors. This situation also may require modifications to processes and criteria that have been used for the existing fleet of plants to assess external hazards and screen them from consideration as providing negligible contributions to PRA results.
- It is postulated that EoCs may become the dominant potential human actions that can impact safety for ARs. Therefore, the analysis of these types of errors will become more important in the assessment of plant risk than it is for the current fleet of LWRs. The current HRA tools do not treat these errors adequately.

- Industry policy organization: The incomplete state of knowledge for ARs will be a challenge for both the regulator and the licensees. Advanced designs are likely to have different drivers of risk than the existing fleet of reactors. Also, there currently is not a large repository of relevant OE for most AR designs. These factors are likely to impact both development and interpretation of PRAs.
- Open discussion sessions:
  - How should the future fleet be prepared to maximize available data? A fundamental question is, what data are needed, and what are the plans for their use?
  - Due to the diverse reactor types undergoing development, what is the best/most cost-effective approach to develop and qualify physics tools (core physics, T-H, and accident progression phenomenology) to support AR licensing and risk analyses? In particular, should industry support development of MELCOR[4] and use it as the accident analysis code suite for NLWR ARs (rather than supporting independent development of industry tools)?
  - Is the historical progression of test reactor to demonstration reactor to commercial reactor the correct approach to accomplish the maturation, licensing, and deployment of ARs? If this approach is pursued, how can the program be structured to obtain the greatest possible amount of information useful for licensing and operation of commercial reactors? From the perspective of plant safety and risk analysis, this is important due to 1) the prominent role of the PRA model/results in the licensing process in the United States and 2) the need to obtain data to characterize and reduce uncertainties related to these first-of-a-kind reactor designs.

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[4] MELCOR is an engineering-level computer code developed by Sandia National Laboratories for the U.S. NRC to model the progression of severe accidents in NPPs. Information is available at https://energy.sandia.gov/programs/nuclear-energy/nuclear-energy-safety-security/melcor/.

– A key objective for many SMRs and NLWR ARs is to reduce the 10-mile EPZ used for licensing the current fleet of LWRs. In LWR licensing, a fundamental assumption is that the entire inventory of fission products ("source term") is available for release to the environment. This assumption could present a challenge to reducing EPZ size, particularly for sites with multiple units potentially subject to common cause initiators, such as extreme external hazards. As indicated previously, the characterization of very low frequency external hazards and their impact on advanced nuclear plants represents a significant challenge. Additionally, use of PRA models has concentrated on assessment of risks to a single unit. Although significant research into assessment of multi-unit risk has been performed, there is currently no consensus on how these analyses should be performed or applied.

– What lessons can be learned from industry LWR experience in use of alternative treatments (10 CFR 50.69[5]), and how can this experience be translated to ARs?

– Passive SSCs generally do not fail catastrophically but are more prone to gradual degradation over time (e.g., due to fouling of passive heat transfer systems); thus, implementation of effective monitoring and diagnostics to detect and address these mechanisms will be more important than it has been for the existing fleet of LWRs.

– Since ARs are being designed to be simpler to operate, with far fewer required operator actions, how can risk methods and tools be used to inform control room staffing requirements?

– From a business perspective, the global market will drive licensing and quality requirements, and thus the subsequent availability of SSCs that will be used in advanced plants. This global supply chain is expected to supply to ISO standards and likely will not be willing to meet different standards (e.g., NQA-1[6]) applied in only one market (e.g., the United States). Impacts of this issue on SSC reliability, availability, and performance may affect risk assessments.

- EPRI presentations:

– RIM was discussed as an example of a new risk-informed application for ARs. It was developed to support detection and management of materials degradation, activities that will support the justification of the SSC reliability and failure rate data used in the PRA models.

  - o RIM provides a comprehensive program to define, evaluate, and implement strategies to ensure that reliability targets for SSCs are defined, achieved, and maintained throughout the plant lifetime.
  - o The approach is included as a requirement in the American Society of Mechanical Engineers (ASME) Section XI Division II code.

[5] 10 CFR 50.69, Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors.

[6] ASME NQA-1 (Nuclear Quality Assurance-1) is an industry consensus standard created and maintained by ASME.

- EPRI is developing an approach to deploy digital I&C systems in ARs [57]. The results of this approach will help to identify areas needing additional research to properly account for their impact on risk.
  - The approach applies systems engineering methods to improve the design of digital I&C systems.
  - The process identifies emergent behavior and potential issues/errors with digital I&C systems using HAZCADS, which integrates the Systems Theoretic Process Analysis (STPA) method to identify unsafe control actions (UCAs).
  - The approach determines risks for system "misbehavior" and uses bounding risk assessments to assign risk reduction targets for each UCA. UCAs produce direct impacts on plant equipment and thus can be directly mapped to failures modeled in the plant PRA.
  - The approach then identifies and establishes control methods (technical or administrative) to minimize the identified risks.
- Since ARs will employ significantly more automation than the existing fleet of LWRs, the assessment of human actions (their application and likelihood) may be different from what has become standard for PRAs of existing plants. The driving questions are:
  - How applicable are existing HRA methods to the digital environment?
  - Can the HRA methods be updated to assess digital systems and operator actions using them?
  - Due to the increased use of automation, it is postulated that EoCs could become a more important (potentially the dominant) source of human error. Is it possible that other tools (e.g., STPA) may be better suited to examine EoCs and eliminate them in the design process?
- EPRI recently published an initial review of ERM and risk-informed decision making [53].
- Key issues identified for risk-modeling tools included the following:
  - What changes to PRA methods and tools would be needed to support risk metrics that are more appropriate for ARs (i.e., metrics other than CDF and LERF, as specified in the ASME/ANS Non-LWR PRA Standard)?
  - What changes in computational accuracy are needed to facilitate AR PRAs?
  - What changes are needed to improve the characterization of uncertainty? Examples of uncertainties that will need to be evaluated for ARs include 1) very low frequency external hazards, which are expected to be a significant contributor to PRA results (at the currently reported levels, the uncertainties of external hazards will likely be far larger than those of internal events), and 2) uncertainties in the modeling of passive system response, including the potential for change in performance over time (due to aging, corrosion, or other fouling mechanisms).

3.3 Research Roadmap

As described earlier, a number of open research questions and needs must be addressed to achieve the objective of licensing and deploying ARs. Prioritization of these needs and allocation of resources to address them are expected to evolve as the AR designs mature, licensing activities progress, and plants are constructed, undergo commissioning, and commence commercial operation. From the literature review, stakeholder interviews, and the AR workshop, a compendium of potential research topics was developed. These issues can be grouped into the following broad categories:

- Methods to assess system reliability and overall risk impacts, including impacts of external events
- Methods to assess human reliability and its impact on AR safety
- Methods to evaluate long-duration events and potentially unique end states
- Methods to address risk using a broader and more integrated approach (e.g., enterprise risk)
- Methods to address plant and component data issues, primarily those necessary to support physics/phenomena analysis (e.g., success criteria definition) and system reliability analyses

These topics provide a useful system to classify and prioritize specific research activities for risk assessments of ARs. From these results, specific research topics to address the identified needs were postulated; for each topic, a qualitative importance/priority was assigned. This assessment characterized the priority for inclusion in the EPRI research portfolio to support ARs into one of three categories: high, medium, or low. A higher priority is assigned to topics that have greater potential impacts on necessary risk modeling capabilities, greater potential impacts on critical design and licensing decisions, and consistent interest from EPRI members and other industry stakeholders. Additionally, the research activities were characterized into one of two prioritization time frames, short term or long term, based on the perceived level of urgency to support AR licensing and deployment schedules and the anticipated level of technical difficulty/cost of the research.

Analysis of data provided from the AR stakeholder community identified 19 specific research tasks, which are characterized according to the scheme indicated above. These are indicated by topic in the following tables.

Table 3-1
Methods to assess system reliability and overall risk impacts

ID	Issue	Importance	Time Frame
3-1-1	Develop a generically applicable approach to evaluate passive system reliability. (Potential starting points include previous EPRI research [31, 32] and RMPS processes developed by Nuclear Emergency Agency [NEA]/Committee on the Safety of Nuclear Installations [CSNI] [55, 56].)	High	Short
3-1-2	Develop methods to identify appropriate risk metrics applicable to monitoring and management of risk for NLWR AR designs, including assessment and monitoring of level of DID associated with various plant configurations.	High	Short
3-1-3	Develop methods to evaluate digital I&C systems, including cyber security, in risk assessments. (Potential starting point is EPRI report describing demonstration of systems engineering approach [57].)	High	Short
3-1-4	Develop methods and tools to efficiently assess and characterize the impacts of very low frequency external hazards (and hazard combinations) on AR risk. Some elements related to the evaluation of the impacts of external hazards may warrant short-term research activities, which are reflected in the near-term research plan discussed below.	High	Long
3-1-5	Develop an approach to assess and manage risk during planned and unplanned changes in plant operating configuration (i.e., AR configuration risk management programs).	Med	Long
3-1-6	Develop methods and tools to evaluate event risks to populate F-C results in a systematic manner that requires less time and resources.	Low	Long

Table 3-2
Methods to assess human reliability and its impact on advanced reactor safety

ID	Issue	Importance	Time Frame
3-2-1	Evaluate applicability of existing HRA methods within the digital environment and identify/prioritize gaps. (Starting point is EPRI report 3002018392 [58].)	High	Short
3-2-2	Evaluate the impact of advanced plant designs and increased automation on the importance of EoCs as a source of human error for ARs.	Med	Short

Table 3-3
Methods to evaluate long-duration events and potentially unique end states

ID	Issue	Importance	Time Frame
3-3-1	Define the characteristics associated with achieving a "safe stable end state" and their impacts on plant PRA evaluations and risk-informed decision making.	Med	Short
3-3-2	Develop methods and tools to identify and assess long-duration sequences that can impact plant risk, including methods to characterize and evaluate uncertainties. A significant element of this issue relates to assessment of human actions and their effectiveness (HRA) related to long-duration events.	Med	Long

Table 3-4
Methods to address risk using a broader and more integrated approach

ID	Issue	Importance	Time Frame
3-4-1	Develop methods to measure and manage project risk (with particular focus on licensing and plant construction).	High	Short
3-4-2	Develop methods to assess and manage plant economic risk throughout the life cycle (design, construction, commissioning, and operation).	Low	Long
3-4-3	Develop guidance to select the most advantageous licensing approach (applicable to the United States where various options exist).	Low	Long

Table 3-5
Methods to address plant and component data issues

ID	Issue	Importance	Time Frame
3-5-1	Develop compendium/prioritization of data needs to support AR licensing and operational decision making.	High	Short
3-5-2	Evaluate potential sources of component reliability data for specific AR designs (these may be different for different NLWR technologies).	High	Short
3-5-3	Identify data needs to support safety analyses for licensing and PRA success criteria.	High	Short
3-5-4	Specify data needs related to use of high-assay low-enriched uranium (HALEU) fuel in ARs (e.g., source term, fission product behavior during accident conditions).	High	Long
3-5-5	Develop plans/approaches to collect and assess operational data on system operational performance and reliability for ARs.	Med	Long
3-5-6	Develop plans/approaches to collect and assess operational data on human factors/errors for ARs.	Med	Long

Based on stakeholder inputs regarding where additional research would be beneficial, the majority of topics relate to the development of methods to assess system reliability (Table 3.1) and data specific to ARs (Table 3.5). These two categories contain 12 of the 19 potential research areas (slightly more than 60%). Stakeholder feedback also identified 10 of the 19 potential research activities (slightly more than 50%) as having high priority, of which 8 (80% of those identified as high priority and 42% of the total) are identified as needing resolution in the short term. This characterization is reflective of the objective to license and deploy multiple AR designs within the next several years.

The issue related to human EoCs was ranked as medium priority based on input from EPRI members. However, this issue has been identified as potentially significant by regulatory authorities and thus could prove an impediment to the efficient and timely regulatory review of license submittals. Hence it is treated here as a high-priority, short-time frame issue to consider for new EPRI research, for an interim total of nine such issues. Additionally, the potential impact of external hazards was characterized as a long-term issue, but it has elements that may require development of methods needed in the short term, particularly methods to support population of F-C curves to support plant licensing. Therefore, a near-term task to identify specific needs to support evaluation of very low frequency external hazards also is included in the development of a near-term research and development program, bringing the total number of issues under further consideration to 10.

Because 10 issues are more than could reasonably be integrated concurrently in the very short term into the EPRI research portfolio, the items were further assessed to identify those on which new research could make substantial progress in the near future. The first step in this assessment was to identify activities for which established research activities are currently in process or which would require broader objectives than those typically addressed in a plant PRA. This evaluation identified the following activities that met this condition:

- Develop methods to evaluate digital I&C systems, including cyber security, in risk assessments.

  Research into this area currently is being conducted by EPRI and other stakeholders to support both ARs and upgrades to the existing fleet of plants. Therefore, a new research task specific to ARs on this topic would not provide sufficient value compared to other potential research tasks.

- Evaluate applicability of existing HRA methods within the digital environment and identify/prioritize gaps.

  Research into this area currently is being conducted by EPRI and other stakeholders to support ARs as well as upgrades to the existing fleet of plants. Therefore, a new research task specific to ARs on this topic would not provide sufficient value compared to other potential research tasks.

- Develop methods to measure and manage project risk.

  Initial research into this area has been conducted by EPRI. Additionally, there is a wide range of literature related to improvements in measuring and managing business risk for large capital projects. Therefore, a new research task specific to ARs on this topic would not provide sufficient value compared to other potential research tasks.

- Evaluate potential sources of component reliability data for specific AR designs.

  Because there is great diversity in the AR designs currently being developed and the characteristics related to equipment reliability will be dependent on the specific design, this issue may best be addressed on a case-by-case basis by the individual reactor vendors. Therefore, a comprehensive research task would not provide sufficient value compared to other potential research tasks.

For these activities, which do not require new research tasks due to ongoing research activities, additional coordination between the different organizations representing the current nuclear fleet and the AR future fleet will be pursued. Note that this assessment reduces the number of candidate high-priority/short-time frame research activities from 10 to 6. In addition, the two remaining data issues are closely related and can be combined into a single research activity. This results in a final list of five proposed AR research activities that should be initiated with higher priority:

- Develop a generically applicable approach to evaluate passive system reliability. The major focus of this task will be to develop approaches to support long-term assessment and maintenance of margins for these systems, approaches that enhance the efficiency and reduce the time and effort required to perform the analyses.
- Develop methods to identify appropriate risk metrics applicable to monitoring and management of risk for NLWR AR designs. This task will include identification of approaches to assess and monitor DID associated with various plant configurations for these reactors.
- Identify specific actions and research needs to assess the impact of very low frequency external hazards (including combined hazards) that are needed to support realistic risk evaluations for plant licensing.
- Evaluate the impact of advanced plant designs and increased automation on the importance of EoCs as a source of human error for ARs. This task will focus on development of methods to identify and characterize potential EoCs that could impact AR risk and safety as well as methods to characterize the related human performance issues and integrate them into the plant PRA.
- Identify and prioritize data needs related to plant safety and risk assessments that are necessary to support efficient and cost-effective licensing of ARs. This research activity will focus on identification of data and relevant characteristics necessary to support plant safety analysis (including anticipated transients, design basis accidents, and beyond design basis accidents) and plant risk assessments (i.e., plant PRAs) that will be used for plant licensing submittals and operational decision making.

These five research areas comprise high-priority, short-term issues related to AR risk and safety that require new research to support the efficient and cost-effective licensing of advanced NPPs. Although these areas were selected from the primary viewpoint of the framework and requirements for plant licensing in the United States, the development of methods to address them will provide benefits to support licensing and deployment of ARs throughout the world. Based on the research activities identified above, EPRI is continuing to engage with AR stakeholders to develop detailed research plans and schedules and incorporate them into the EPRI research and development portfolio.

3.4 Linking to EPRI/NEI Advanced Reactor Roadmap

Recently EPRI and NEI published an Advanced Reactor Roadmap [59] to provide a compendium of prioritized actions and timelines to enable AR deployments in the United States and Canada. The Advanced Reactor Roadmap identifies a broad range of activities (intended to be comprehensive) necessary to achieve widespread deployment of ARs by the mid-2030s. The Advanced Reactor Roadmap is constructed at a high level. It identifies and prioritizes issues (classifying them into two tiers); however, detailed recommendations and schedules for solutions are the responsibility of the individual stakeholders (e.g., reactor vendors, plant owner/operators, and regulatory authorities), who will develop and execute them.

The research activities described in this report generally support several aspects of the Advanced Reactor Roadmap, which is organized into 46 identified actions within 13 strategic elements within three pillars. Though most actions are not owned by EPRI, the technical basis provided by improved risk analysis methods and tools through this project should be an important contributor to the following pillars, strategic elements, and actions from the Advanced Reactor Roadmap.

- Regulatory efficiency
  - Licensing
    - Develop recommendations for enhancements to licensing processes (owned by NEI and CNA)
    - Develop industry recommendations for NRC guidance on operator staffing (owned by NEI and the Canadian Nuclear Association [CNA])
  - Environmental
    - Develop technical input to siting criteria (owned by NEI)
- Technology readiness
  - Plant/SSC design
    - Develop and qualify analytical tools for AR design (owned by EPRI)
    - Develop guide on leveraging legacy reactor experience (owned by EPRI)
  - Nuclear beyond electricity (NBE)
    - Establish decoupling framework for NBE users (owned by EPRI, NEI, CNA, AR developers, and owners)

- Codes and standards
  - Demonstrate risk-informed and performance-based approach (owned by the American Nuclear Society [ANS], ASME, and CSA with NEI, CNA, and AR vendors)

- Project execution
  - Initial operations and maintenance
    - Reduce operating and maintenance costs to a level similar to that at other thermal plants (multiple actions owned by EPRI, AR vendors, and the Institute of Nuclear Power Operations [INPO])

Mapping of this report's recommended R&D activities to the Advanced Reactor Roadmap identifies where each research activity can contribute to addressing the issue(s) identified in the Advanced Reactor Roadmap, though the activity may not provide a complete solution by itself. The five high-priority research activities identified in this report support one or more of the following Advanced Reactor Roadmap actions.

- Develop a generically applicable approach to evaluate passive system reliability.

  Advanced Reactor Roadmap actions supported:

  - Develop and Qualify Analytical Tools for Advanced Reactor Designs: The approaches for the evaluation of passive system reliability supported by this R&D task will support risk-informed design approaches common to many AR design processes.
  - Develop Enhancements to Licensing Process: This R&D task will result in guidance and methods to consistently evaluate the performance of passive systems in plant risk and safety analyses, which will generally support more efficient regulatory review and approval of safety analyses.
  - Demonstrate Risk-Informed and Performance Based Approach: Because passive system performance and reliability is a critical element in plant safety and risk, this R&D activity can generally support demonstrating a risk-informed and performance-based framework.
  - Reduce Operating and Maintenance Costs to a Level Similar to Other Thermal Plants: Because the use of passive systems as the primary means of ensuring plant safety in response to accidents and transients is a relatively new concept for commercial reactors, the reliability of these systems is a critical element in assuring adequate nuclear safety. One aspect of this R&D activity will address potential degradation in system reliability over time (e.g., due to buildup of corrosion products that could impact heat transfer capabilities). The outcomes can leverage these safety features to identify efficient inspection and maintenance activities while maintaining a high level of nuclear safety over the plant operational life.

- Develop methods to identify appropriate risk metrics applicable to monitoring and management of risk for NLWR AR designs.

  Advanced Reactor Roadmap actions supported:

  – Develop Technical Input to Siting Criteria: In the envisioned AR licensing framework (as described in NEI 18-04 [8]), assessment of plant risk (via characterization by F-C curves) provides a primary set of criteria in the licensing process. This R&D task will develop applicable risk metrics that can provide technical bases for site-specific licensing decisions.

  – Demonstrate Risk-Informed and Performance Based Approach: Because the capability to effectively assess and monitor risk is fundamental to both plant owner/operator responsibilities and regulatory review/oversight in a risk-informed, performance-based framework, this R&D activity can generally support demonstrating this framework.

- Identify specific actions and research needs to assess the impact of very low frequency external hazards.

  Advanced Reactor Roadmap actions supported:

  – Develop Enhancements to Licensing Process: This R&D task will result in guidance and methods to evaluate the impact of external hazards in plant risk analyses, which will generally support more efficient regulatory review and approval of safety analyses.

  – Establish Decoupling Framework for Nuclear Beyond Electricity (NBE) Users: The plant footprint for NBE applications (such as providing high-temperature process heat to industrial facilities) may be different than that of the existing fleet of reactors used exclusively for electric power production. This is due to potential interactions and interdependencies among the reactor and associated NBE applications. One aspect of this R&D task will develop an approach and guidance to assess these interactions/dependencies in the context of external hazards and their potential impact on reactor risk and safety.

  – Demonstrate Risk-Informed and Performance Based Approach: Because external hazards may be a significant contributor to risks associated with ARs, this R&D activity can generally support demonstrating a risk-informed and performance-based framework.

- Evaluate the impact of advanced plant designs and increased automation on the importance of EoCs as a source of human error for ARs.

  Advanced Reactor Roadmap actions supported:

  – Develop Enhancements to Licensing Process: This R&D task will result in guidance and methods to support comprehensive HRAs for ARs, which will generally support more efficient regulatory review and approval of safety analyses.

  – Develop Industry Recommendations for Regulatory (NRC/CNSC) Guidance on Operator Staffing: Results from this R&D activity may be used to support the risk-informed technical basis related to operator staffing levels.

- Identify and prioritize data needs related to plant safety and risk assessments that are necessary to support efficient and cost-effective licensing of ARs.

Advanced Reactor Roadmap actions supported:

- Develop Guide on Leveraging Legacy Reactor Experience: In addition to design-specific data, legacy OE may also provide support for safety analysis and risk assessments. The outcome of this R&D activity will support the use of legacy data by identifying and prioritizing the overall data needs.
- Provide Joint Recommendations to North American Regulators (NRC and CNSC) on Regulatory Alignment: A common element of reactor licensing will be the use of experimental and (when available) operational data to substantiate and qualify the plant licensing basis, including both traditional plant safety analyses and risk assessments. The outcome of this R&D activity will generally support the identification of the data needed for these analyses for licensing of ARs. This task should be performed, to the extent practicable, to support licensing beyond the United States and Canada.

4
CONCLUSIONS AND RECOMMENDATIONS

Many of the features that classify new reactor designs as “advanced” present challenges for the approaches that currently are used to perform PRA of plant safety for the current fleet of LWRs. Many ARs are designed to include less inventory of radioactive materials, employ more stable fuel forms, and possess higher system thermal capacities with longer thermal (time) constants, as well as employing passive safety features that rely on natural forces to provide enhanced safety when compared to the current generation of operating plants. In addition, many ARs employ different materials and physical processes. Finally, the number of diverse reactor designs undergoing active development, many with different combinations of advanced features, results in a diverse set of considerations for methods and tools for assessing plant safety and risk.

The objective of this report is to identify and prioritize technical issues related to the conduct of risk assessments of advanced nuclear plants. Greater consideration/priority was given to identifying and addressing gaps important to near-term licensing actions for ARs. The research resulted in identification and prioritization of areas where research is needed to develop, validate, and disseminate the necessary data and methods to better analyze plant risk for AR designs.

Based on information provided by the community of AR stakeholders, a proposed set of five research activities that should be initiated with higher priority was identified:

- Develop a generically applicable approach to evaluate passive system reliability. The major focus of this task will be to develop approaches that support long-term assessment and maintenance of margins for these systems, approaches that enhance the efficiency and reduce the time and effort required to perform the analyses prescribed in those previously developed to address this issue.
- Develop methods to identify appropriate risk metrics applicable to monitoring and management risk for NLWR AR designs. This task will include identification of approaches to assess and monitor DID associated with various plant configurations for these reactors.
- Identify specific actions and research needs to assess the impact of very low frequency external hazards (including combined hazards) that are needed to support realistic risk evaluations for plant licensing.
- Evaluate the impact of advanced plant designs and increased automation on the importance of EoCs as a source of human error for ARs. This task will focus on development of methods to identify and characterize potential EoCs that could impact AR risk and safety as well as methods to characterize the related human performance issues and integrate them into the plant PRA.

- Identify and prioritize data needs related to plant safety and risk assessments that are necessary to support efficient and cost-effective licensing of ARs. This research activity will focus on identification of data and relevant characteristics necessary to support plant safety analysis (including anticipated transients, design basis accidents, and beyond design basis accidents) and plant risk assessments (i.e., plant PRAs) that will be used for plant licensing submittals and operational decision making.

These five research areas comprise high-priority, short-term issues related to AR risk and safety that require new research to support the efficient and cost-effective licensing of advanced NPPs. Although these areas were selected from the primary viewpoint of the framework and requirements for plant licensing in the United States, development of methods to address them will provide benefits to support licensing and deployment of ARs throughout the world. Based on the research activities identified above, EPRI plans to conduct further engagements with AR stakeholders and to develop detailed research plans and schedules, which will be incorporated into the EPRI research and development portfolio.

A conceptual roadmap for the research tasks identified in this report is provided in Figure 4-1. This roadmap identifies activities that have been completed, activities currently under progress, and activities expected to be undertaken in the future. It is intended that this roadmap will undergo additional development (e.g., identify specific dates for completion) and mature as the licensing and deployment of ARs proceed.

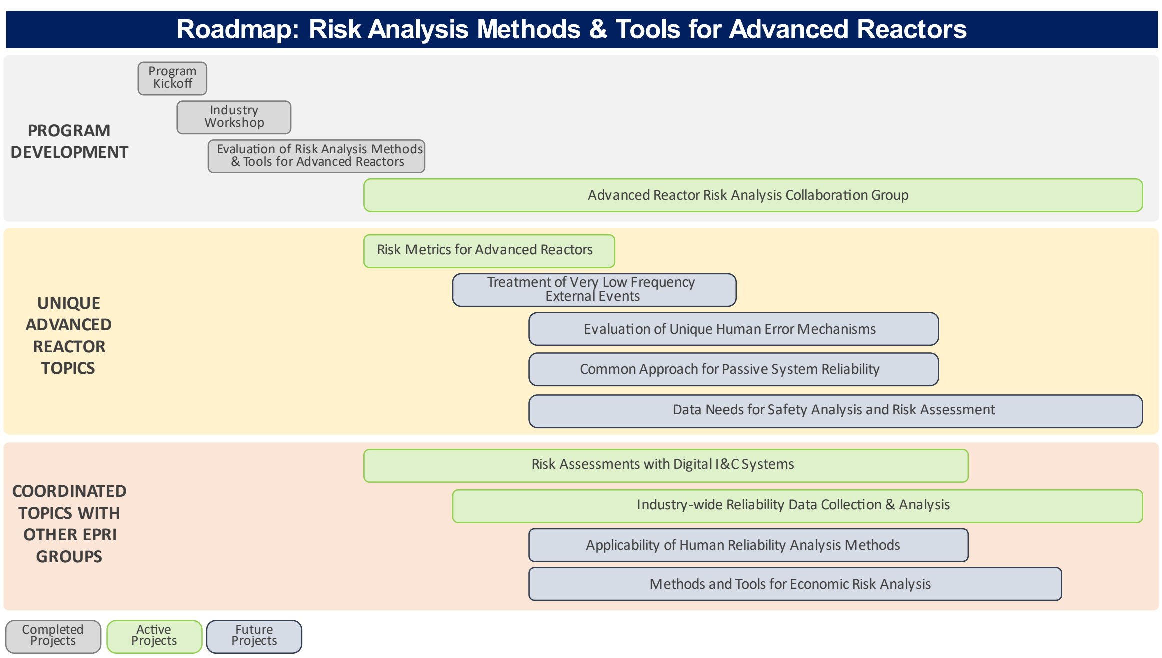

Figure 4-1
Conceptual AR PRA research roadmap

As discussed in Section 3.4, EPRI and NEI recently partnered to develop the Advanced Reactor Roadmap (Phase 1: North America) [59] to identify critical strategies and actions necessary to support the development and deployment of ARs. The topics identified in this report for further research will work together to improve the capabilities of ARs to estimate risks and implement risk-informed applications, which will generally support several aspects of the Advanced Reactor Roadmap, as identified. Such activities also are expected to evolve as the design, licensing, and deployment of ARs progress. The research activities identified in this report will mature to meet the evolving needs of AR deployments, and additional research activities may be identified to address emerging stakeholder needs.

5
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