An Overview of Current Risk-Informed Decision-Making Research Activities at the US Nuclear Regulatory Commission

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Abstract: The United States Nuclear Regulatory Commission (NRC) is responsible for regulatory oversight of the United States' civilian use of radioactive materials. The NRC's mission focuses on ensuring public health and safety and protecting the environment. In support of these mission objectives, the NRC's Office of Nuclear Regulatory Research (RES) conducts a variety of activities to support the increased use of risk information in the regulatory process. Consistent with the vision embraced by the NRC's Policy Statement on the Probabilistic Risk Assessment (PRA), the NRC is developing and maintaining data, tools, and methods to support the increased use of risk information in licensing, oversight, and regulatory programs. Much of this work is driven by emerging changes in the US nuclear fleet (e.g., increased use of digital instrumentation and control, deployment of new fuel designs capable of supporting longer fuel cycles, and increasing recognition of the ability of risk-informed processes to allow operational flexibility while maintaining safety) and high interest in advanced reactor designs. Notable examples of recent work include development of enhanced tools to support risk assessment of high energy arcing faults, consensus standards activities for PRA technology supporting light water and non-light water advanced reactors, data collection and parameter estimation, maintenance and enhancement of independent Standardized Plant Analysis Risk (SPAR) PRA models, application of the Integrated Human Event Analysis System (IDHEAS) human reliability method to events and condition assessment, and progress toward completion of the Level 3 PRA project. In addition, research activities continue to be performed to examine advanced PRA topics such as dynamic PRA and application of PRA licensing of non-light water reactors. This paper will discuss the role of RES in supporting the NRC's mission, provide a summary of these and other key risk-related research activities, and highlight planned future areas of research focus. Challenges, opportunities, application examples and guidance examples are also highlighted for each application area.

Keywords: PRA, Risk Informed Decision Making, Research, Fire PRA

1. INTRODUCTION

As described in the NRC's Strategic Plan [1], a main function of the NRC is to conduct research and risk performance assessments to support regulatory decisions. This function supports several strategies outlined in the strategic plan, including the promotion of risk-informed decision-making; assessing the potential of increased risk due to climate change; maintain and further risk-inform the current regulatory framework; development of effective communication strategies to explain how risk and uncertainty are addressed and considered in the decision-making process; and continue to achieve mission excellence as a modern risk-informed regulator. The Office of Nuclear Regulatory Research plays a critical role in meetings these strategies through a variety of activities described in in the Research Prospectus for FY2022 through FY2024 [2]. Within the NRC's Office of Nuclear Regulatory Research, the Division of Risk Analysis supports the agency Strategic Plan by developing, recommending, planning, and managing research programs relating to probabilistic risk assessments (PRA) and human reliability analysis. In addition, U.S. operational safety data and reliability information is routinely assessed to determine risk-significant insights and trends. In broad terms, the NRC's PRA and risk research activities can be organized into four functional areas: (1) support the reactor oversight and operating experience programs; (2) remove obstacles to the implementation of risk-informed regulation; (3) expand the use of PRA and risk evaluation to encompass advanced reactor designs, and (4) support continuous advancement in the PRA state-of-the-art and state-of-practice [3]. This paper describes key research activities in each of these areas.

2. Support the Reactor Oversight and Operating Experience Program

As discussed in NRC Inspection Manual Chapter 0308 [4], in April of 2000 the NRC implemented a revised oversight process that was more risk-informed, objective, and predictable. The revised oversight was designed to maintain safety, increase openness, make NRC activities and decisions more efficient, effective, and realistic, and reduce unnecessary regulatory burden. Key features of this process included an increased reliance on NRC risk tools such as Standardized Plant Analysis Risk (SPAR) models; the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) probabilistic risk assessment computer code; and data collection and analysis activities. In addition, the NRC has continually improved the risk tools supporting the Accident Sequence Precursor (ASP) program, which was established in 1979 in response to recommendation contained in the WASH-1400 Risk Assessment Review Group Report [5].

2.1. SPAR and SAPHIRE

The SPAR models are plant-specific, NRC-developed, probabilistic risk assessment models that use standardized modeling conventions and data [6]. The standardized approaches used to develop event trees, fault trees, basic events, data, and naming conventions improve the efficiency of agency risk analysts in using the SPAR models, which represent a diverse collection of commercial nuclear power plant designs. In addition, the standardization of the models makes activities such as data updates, generating risk insight reports, and analysis of conditions across multiple SPAR models more efficient. Although the parameters used in the SPAR models utilize industry average operating experience data (e.g., see https://nrcoe.inl.gov/), the models are benchmarked against licensee-developed PRAs when possible to ensure that any differences in risk results due to the use of standard modeling conventions or industry averaged data are identified and understood. Because the SPAR models are specifically developed to support initiating event and unavailability condition assessments, the models include several features, including detailed offsite power reliability, power recovery, and common cause failure models, that improve the efficiency of model adjustments to support analyses typically encountered by agency risk analysts. The NRC currently maintains 67 SPAR models, representing all currently operating U.S. nuclear plants. All of these models include Level 1 PRA modeling (core damage frequency) for internal hazards, high winds, and seismic hazards. Twenty-three of these operating reactor models include fire PRA modelling, based either on the early fire PRA studies or more recent models used to support risk-informed fire protection programs (NFPA 805). The staff also maintains six SPAR models for new reactors designs (e.g., ABWR, US EPR, APWR). In addition, the staff is enhancing SPAR model capabilities to increase the number of models updated each year from six models to twelve, expand fire modelling capabilities to more models, and develop models for new reactors. To further improve the ability to communicate risk insights obtained from the SPAR models, a SPAR model dashboard has been provided to agency staff to increase the access and use to key risk insights. The staff is also completing a pilot application to expand the usage of the Integrated Human Event Analysis Systems for Event and Condition Assessment (IDHEAS-ECA) [7] by building a knowledge base of documented human failure event (HFE) evaluations. The selected HFEs are identified through a review of the SPAR models considering risk significance, number of associated models, and impact on typical event and condition assessments (ECAs) encountered by NRC risk analysts. Other goals of this pilot study include identifying areas for method and/or guidance improvement; enhancing the NRC's in-house experience in applying IDHEAS-ECA method; better understanding of the impact of IDHEAS-ECA on the base SPAR model results; assessing longer term options for updating SPAR model Human Reliability Analysis modelling and communicating the results of the pilot study to the agency risk community.

The SPAR models are run using the SAPHIRE computer code. In addition to the capabilities associated with any modern fault tree/event tree PRA code, the SAPHIRE code includes additional features that support ECA activities, including uncertainty propagation, common cause failure modelling, ability to calculate changes in core damage frequency, and several user workspaces that can simplify use of the code for more straightforward analyses. The SAPHIRE code is also used by other users and organizations, including U.S. federal agencies, such as the National Aeronautics and Space Administration (NASA). The SAPHIRE code is maintained under a quality assurance program that complies with NUREG/BR-0167 [8]. The staff strives to continually improve the capabilities in SAPHIRE, with recent activities focused on improvements to the computational solver to improve computational speed, enhancement of large early release calculational capabilities, improvements to success path quantification to enhance the handling of high failure probabilities, and development of a cloud-based SAPHIRE code that can better leverage the high performance computing resources while maintaining computer security requirements. Other future work includes incorporating tools in SAPHIRE for assessing risk informed technical specification completion times [9] and uncertainty quantification.

2.2. Accident Sequence Precursor Program

The ASP program reviews reactor operational events to determine their risk significance. These events are broadly classified into two categories: (1) a degraded plant condition characterized by the unavailability or degradation of systems, structures, or components in the absence of a specific initiating event, and (2) the occurrence of an initiating event (e.g., a reactor trip) with or without any other degraded plant conditions [10]. Degraded plant conditions are assessed as a change in core damage probability over the time period that the degraded condition existed (ΔCDP) and initiating events are assessed using a conditional core damage probability (CCDP) for the initiator. The ASP program defines a "precursor" as a degraded condition with a ΔCDP greater than or equal to 10^{-6} or an initiating event with a CCDP greater than 10^{-6} or the CCDP of a non-recoverable loss of feedwater or condenser heat sink (whichever is larger).

The ASP program was initiated in 1979 in response to a recommendation contained in NUREG/CR-0400. As noted by the authors of NUREG/CR-0400, "it is important, in our view, that potentially significant sequences and precursors, as they appear, be subjected to the kind of analysis contained in WASH-1400" [5]. One stated motivation for this view was to address uncertainty about the completeness of accident sequences included in the WASH-1400 study. The ASP program objectives include identification of precursors to potential core damage, evaluate operating experience and trends to enhance the regulatory framework as warranted, and assessment of existing agency programs [10]. Additionally, ASP events with a CCDP or a change in core damage probability ΔCDP of greater than or equal to 0.001 are reported to the U.S. Congress as an abnormal occurrence in NUREG-0090 [11].

The ASP program utilizes the SPAR models, SAPHIRE code, and supporting operating experience data to evaluate operational events. The results from ASP program reviews are published annually and made publicly available on NRC ASP public webpage (https://www.nrc.gov/about-nrc/regulatory/research/asp.html). The ASP annual report [12] provides trend information for several categories of precursors, including initiating events, degraded conditions, important precursors (i.e., CCDP/ΔCDP greater than or equal to 10^{-4}), losses of offsite power, emergency diesel generator failures, boiling water reactors and pressurized water reactors. An example of the trending results available in the annual report is shown in Figure 1.

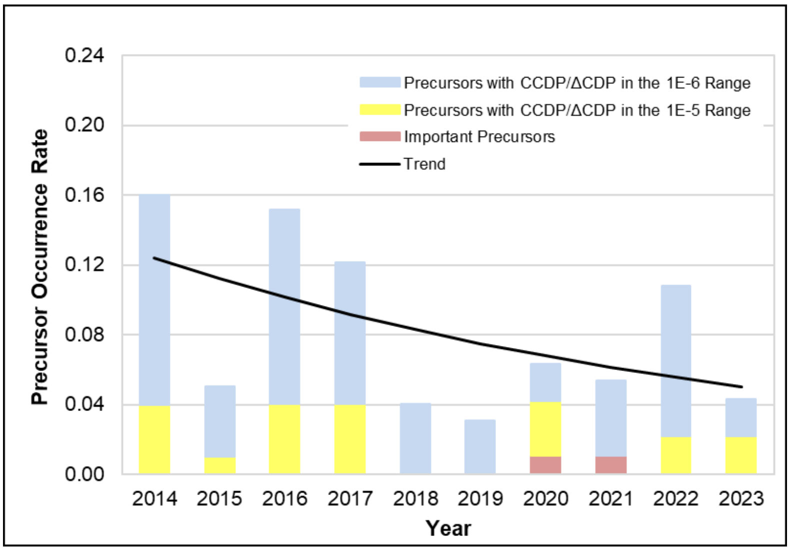

Figure 1. 2023 Historical Occurrence Rate of All Precursors

2.3. Operating Experience Data

A key element of the NRC's PRA Policy Statement [13], is that PRA evaluations should be as realistic as practicable and appropriate supporting data should be publicly available. The NRC addresses this objective through an active operating experience data collection and analysis program. A key input for the NRC's data program is data collected by the Institute for Nuclear Power Operation (INPO) under their Industry Reporting and Information System (IRIS). The NRC, with support from Idaho National Laboratory (INL), codes and collects operating experience data in the Reliability and Availability Data System (RADS). Because both IRIS and RADS contain proprietary data, the NRC and INL remove proprietary information from this data and make industry averaged data (which does not plant identifiers) publicly available through a website (https://nrcoe.inl.gov/). The industry averaged data is analyzed periodically to develop updated reliability parameter estimates for use in SPAR models or other risk assessment studies. In addition, the data is used to develop other products such as initiating events trends and insights; component and system studies; and common cause failure insights. Figure 2 provides an example of the types of insights available from these data studies [14].

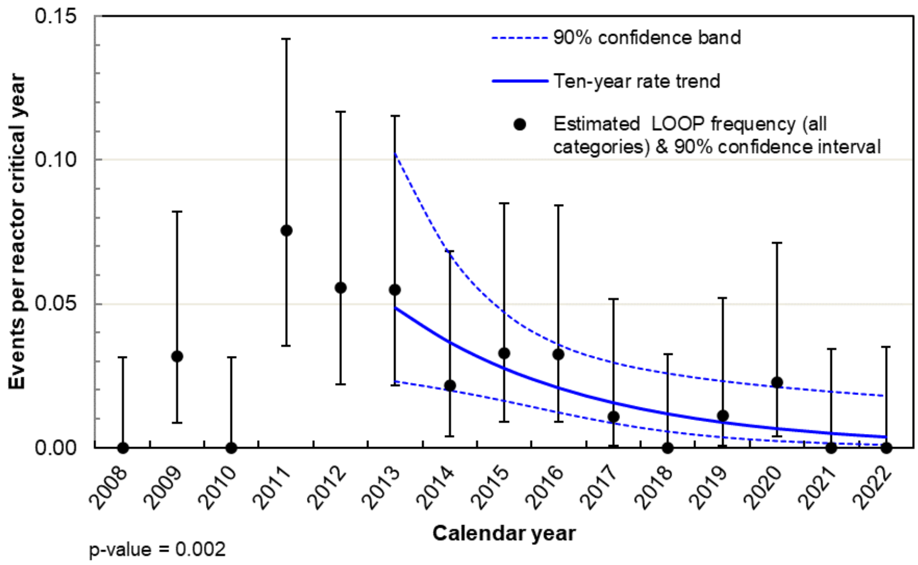

Figure 2. Historical Trend for Estimated LOOP Frequencies

Collectively, this information assists the NRC in meeting its safety mission while making data supporting risk assessments available to the public.

3. Remove Obstacles to the Implementation of Risk-Informed Regulation

One objective of increasing the use of risk-informed decision making at the NRC is to increase efficiency and enhance safety by focusing on issues that have the greatest benefit to public health and safety. It has long been understood that removing barriers to the implementation of risk-informed methods increases staff efficiency, but can also increase the effectiveness of the regulatory process and improve safety. To this end, the NRC has actively been addressing issues that could be barriers to increased use of risk informed methods and approaches. Two examples highlighted in this section are PRA standard activities and the development of improved methods to increase the realism of fire PRAs.

3.1. PRA Standards Activities

U.S. law [15] requires that federal agencies, such as the NRC, emphasize "where possible the use of standards developed by private, consensus organizations." To support implementation of this law, the U.S. Office of Management and Budget (OMB) issued OMB Circular A-119-1, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities" [16]. Circular A-119-1 directs "agencies to use voluntary consensus standards in lieu of government-unique standards except were inconsistent with law or otherwise impractical." Consistent with this guidance, the NRC has worked with the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) to develop and implement consensus standards for the regulatory use of PRA. A significant benefit of the widespread use of consensus standards is increased assurance that PRA models are acceptable for regulatory use.

In 2004, the NRC developed a plan for phased approach to PRA quality[1] [17]. An objective of the phased approach plan was that potential applications of PRA would be implemented in a consistent and predictable

[1] Although the term "PRA quality" has been used extensively in the past, the current term used in the U.S. is "PRA acceptability". These and other related terms are synonymous and refer to the scope, level of detail, conformance with

manner that would promote regulatory stability and efficiency. Key activities highlighted in the phased approach included development of PRA peer review guidance, development of PRA standards by ASME and ANS, and issuance of regulatory guidance by the NRC to implement a process that ensures PRA technical acceptability. The three phases were defined as follows:

- Phase 1 – Application Specific Phase – Focus on specific application areas such as inservice inspection, quality assurance, and allowed outage times. The PRA is commensurate with the application for which it is intended and the role that PRA results play in the integrated decision process. Risk contributions that are not addressed by the PRA are addressed using qualitative analysis, bounding methods, compensatory measures, or by defining the change so that these contributions are not challenged.
- Phase 2 – Application Type Phase – Focus on general application types where the baseline PRA meets applicable consensus standards, and the PRA scope includes all operational modes and initiating events that could substantially change the regulatory decision. Phase 2 is characterized by conformance with applicable standards and completion of a peer review. In this phase, the focus of the staff's PRA review is on (1) those portions of the baseline PRA that were determined by a peer review to not be in conformance with the standard and that are significant to the application and (2) on key assumptions and uncertainties significant to the decision.
- Phase 3 – All-Applications Phase – This phase is similar to Phase 2 but focuses on the PRA having sufficient scope to address all envisioned applications and meeting industry consensus standards. A significant difference between Phases 2 and 3 is that Phase 3 envisioned a one-time NRC staff review of the PRA rather than application specific reviews (as are done in Phase 2).

A central theme for establishing PRA technical adequacy is the interplay between national consensus standards developed by ASME and ANS under the Joint Committee on Nuclear Risk Management[2] (JCNRM), development of staff positions on standards and other areas of regulatory significance, and the peer review process. This framework is highlighted in Regulatory Guide (RG) 1.200 [18] where the combination of NRC staff positions, the use of national consensus standards, and peer reviews to demonstrate conformance with consensus standards form the basis of establishing PRA acceptability. Since the issuance of the Phased Approach to PRA quality in 2004, the NRC has made significant advancements, including issuance and updates of foundational regulatory guidance in RG 1.174 [19], RG 1.200 [18], and NUREG-1855 [20]. As such, the agency has reached the vision outlined in Phase 2, with work being done to support further advancements.

Current NRC priorities for PRA standards development are periodically communicated to the ASME Board on Nuclear Codes and Standards. The most recent communication [21] to ASME, highlighted the following priorities for the NRC:

- ASME/ANS RA-S-1.1-2022: The new edition for the Level 1/Large Early Release Frequency (LERF) PRA standard addresses at-power conditions and all hazards for light water reactors (LWRs). The NRC staff is considering endorsement of this standard, with appropriate clarifications or qualifications, as one of the main elements of next revision to RG 1.200
- ASME ANS RA-S-1.2: The Level 2 PRA standard addresses all operating modes and hazards for LWRs and has been issued for trial use, has undergone pilot applications (one pilot being the NRC Level 3 PRA project discussed in Section 5,1) and is being finalized to be issued as an ANSI standard. The NRC plans to endorse this standard, with appropriate clarifications or qualifications, in the next revision of RG 1.200.
- ASME/ANS RA-S-1.3: The Level 3 PRA standard addresses all operating modes and hazards for LWRs. This standard has been issued for trial use and is currently being piloted (one pilot being the NRC Level 3 PRA project discussed in Section 5.1). The NRC considers completion of this standard to be a lower priority.
- ASME/ANS RA-S-1.4-2021: The non-LWR PRA standard addresses all operating modes, hazards, mechanistic source terms, radiological consequences, and risk integration for design, preconstruction, pre-operation, and operating stages. This standard has been issued as an ANSI standard. This standard has been endorsed with clarifications or qualifications for trial use in RG 1.247 [22].

PRA technical elements, and the plant representation of a PRA (i.e., the PRA representing the as-built and as-operated plant).
[2] See https://www.ans.org/standards/committees/jcnrm/ for more information.

- ASME/ANS RA-S-1.5: The Level 1/Level 2 PRA standard addresses advanced LWRs during preconstruction and preoperational stages. The NRC considers the completion of this standard to be a high priority and plans to endorse, with appropriate clarifications or qualifications, in the next revision to RG 1.200.
- ASME/ANS RA-S-1.6: The Level 1/LERF PRA standard addresses low-power and shutdown conditions for both internal and external hazards for LWRs. The NRC considers the completion of this standard to be a high priority and plans to endorse this standard, with appropriate clarifications or qualifications, in the next revision of RG 1.200.
- ASME/ANS RA-S-1.7: The Level 1/Level 2/Level 3 PRA standard addresses all operating modes and hazards for multi-unit PRA models. The NRC considers the completion of this standard to be a lower priority.

In addition to support for standards development activities through participation on consensus standards committees and working groups and issuance of regulatory guidance updates, the NRC is also developing additional tools to assist in the reviews of risk informed applications. For example, staff is currently developing a database of methods that have been used in nuclear power plant PRAs and a PRA standards database, which will include published and draft PRA standard requirements and comparisons between related requirements from different PRA standards. In addition, the staff is working to update NUREG-2122, “Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking,” [23] and implement other enhancements for communication with external stakeholders.

3.2. Fire PRA

The NRC’s fire protection requirements are contained in Title 10 to the U.S. Code of Federal Regulations (CFR) Part 50.48, “Fire protection. The NRC’s regulations provide two options for meeting this requirement: (1) a deterministic option described under 10 CFR 50.48(b); or (2) a risk-informed, performance-based option provided under 10 CFR 50.48(c) which allows use of the National Fire Protection Association (NFPA) Standard NFPA 805, 2001 Edition [24], with some exceptions. NFPA-805 includes several performance based attributes [25], including: use of measurable or calculable parameters to monitor the system, including facility performance; establishment of objective criteria to assess performance based on risk insights, deterministic analyses, and/or performance history; flexibility in determining how to meet established performance criteria; and a framework to assess the failure to meet performance criterion that ensures that such failures, while undesirable, will not constitute or result in an immediate safety concern. The NRC guidance for implementation of the NFPA-805 option is contained in RG 1.205 [26] and notes that while a fire PRA is not mandatory to transition to NFPA 805, a plant-specific fire PRA enables a licensee to fully realize the safety and cost benefits of NFPA 805 (e.g., licensee self-approval of certain risk-informed changes to the fire protection program). The NRC, in cooperation with the Electric Power Research Institute (EPRI), provided initial fire PRA guidance in NUREG/CR-6850/EPRI 1011989, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Final Report” [27]. Since the issuance of NUREG/CR-6850/EPRI 1011989 in 2005, the NRC has actively continued fire PRA research activities to further increase the realism of fire PRA. Much of this work has been done in cooperation with EPRI under a memorandum of understanding and through international collaborative activities such as the Organization for Economic Cooperation and Development FIRE data project (https://www.oecd-nea.org/jcms/pl_24954/fire-incidents-records-exchange-fire-project) and high energy arcing fault joint projects (e.g., https://www.oecd-nea.org/jcms/pl_24977/high-energy-arcing-fault-events-heaf-project).

Recent fire PRA work has focused on incorporation of lessons learned from international high energy arcing fault testing, augmented with additional insights from an experimental and a joint NRC/EPRI expert working group, into updated guidance that improves the technical basis for evaluating equipment fragility and the zone of influence for these high energy events [28] [29]. The issuance of this updated guidance was a capstone achievement for a multiyear research program that involved experimental programs to investigate electrical arc behaviour, cable fragility, large-scale equipment arcing fault testing, computational fluid dynamics, and operating experience (Figure 3). The updated methods increase fire PRA realism in several areas, including providing consideration of electrical fault clearing times (which influences arc energy), an improved technical basis for zones of influence, improved estimates for cable fragility during HEAF events, and mitigation credit for electrical raceway fire barrier systems.

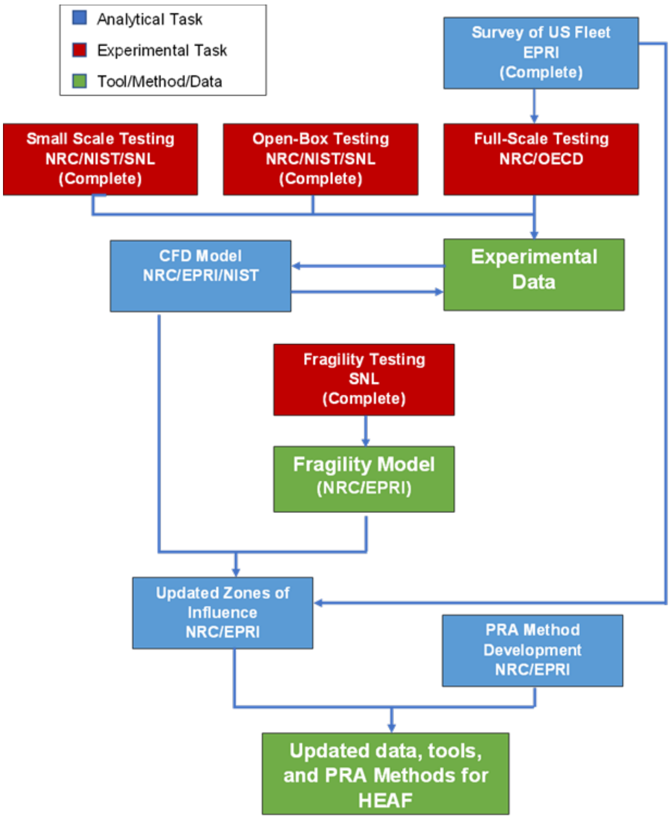

Figure 3. High Energy Arcing Fault Research Program

4. Expand the use of PRA to Encompass Advanced Reactor Designs

The NRC is currently developing and implementing new risk-informed approaches for the licensing on non-LWR reactor designs. Advanced reactors are anticipated to have several advantages over the current LWR nuclear power reactor fleet, including enhanced safety margins and use of simplified, inherent, passive, or other innovative means to accomplish safety and security functions [30]. The NRC is currently developing a new risk-informed, performance-based licensing framework for non-LWRs that allows flexibility and leverages risk insights. This licensing approach, which uses the NEI 18-04 framework [31], is expected to become a key element of a new regulation, 10 CFR 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Commercial Nuclear Plants" [32]. To support deployment of this new framework, ongoing and planned research activities include development of new risk metrics for advanced reactors (to supplement or replace currently used metrics like core damage frequency and large early release frequency), identification of data sources for advanced reactor designs (which may include new components types and design and operating conditions), development of guidance for addressing uncertainties, and activities to build knowledge and experience with PRA modelling challenges unique to advanced reactors.

5. Continuous Advancement in the PRA State-of-the-Art and State-of-Practice

Although generally a smaller portion of the risk-related research portfolio, the NRC conducts several research activities to build future capabilities to advance the state-of-the-art and state-of-practice in PRA technology. Two such programs are highlighted – the multiunit, all modes, all hazards Level 3 PRA project and Future Focused Research.

5.1. Level 3 PRA Study

As noted in SECY 12-0123 [33], the objectives of the ongoing Level 3 PRA project are to reflect technical advancements since the completion NUREG-1150 [34], extract new risk insights to enhance decision-making, enhance the staff's PRA capabilities, and obtain insights into the costs of developing Level 3 PRA models. This project includes numerous aspects not previously included in NUREG-1150, including all modes of operation (power operation and shutdown), all hazard categories (although NUREG-1150 did address internal fires and seismic events for two plants), and all significant radiological sources (core, spent fuel pool, and dry cask storage). The staff has completed the majority of the technical work and has issued several portions of the final report for public comment. The staff anticipates issuing the remaining project reports for public comment in 2024 and 2025.

To date, the project has successfully met many of its objectives. Nearly 100 NRC staff and several U.S. national laboratories and commercial contractors have supported the work. This has enabled NRC staff to build high levels of capabilities for all levels of PRA modelling and has helped to maintain critical core competencies within the NRC contracting community. The study also provided opportunities to pilot streamlined expert elicitation guidance (used to assess intersystem loss of coolant accident frequencies), and pilot applications of the ASME/ANS Level 2 and 3 standards during project peer reviews. The staff has benefited greatly from cooperative assistance from the Pressurized Water Reactor Owners Group, EPRI, Westinghouse, licensee staff from the representative site, and industry staff who supported peer review activities for the study. While estimating the costs of conducting a Level 3 PRA was a stated objective for the project, the need to address other emergent higher priorities, such as the response to the Fukushima Dia-ichi accident, resulted in some delays of certain project milestones. In addition, at times the agency prioritized the training opportunities for staff over a rapid completion schedule. This allowed the agency to build significant technical capability by utilizing agency staff to perform a significant amount of the PRA modelling, thermal hydraulic, and consequence analysis in house.

5.2. Future Focused Research

The NRC's Future Focused Research (FFR) program identified future research needs by tapping into the expertise of the entire NRC staff. The FFR uses a streamlined approach to conduct small scale research projects. The objectives of the program are to improve future capabilities, build foundational knowledge, and address longer term research gaps. A sampling of several risk related FFR projects include the following:

- Dynamic PRA (DPRA) Study (complete) – This study conducted several activities to assess the use of DPRA techniques in risk-informed decision making, including literature surveys, workshops, and an application of a DPRA tool to a reactor application.
- Apply the Licensing Modernization Project Methodology on an Operating Reactor (complete) – This project applied the results are derived from the NRC developed level 1, level 2, and level 3 base case PRA for internal events, internal floods, and external hazards (described in Section 5.1) to the NEI 18-04 [31] Licensee Modernization Project (LMP) frequency-consequence (F-C) curve to both gain experience with the overall LMP methodology and to identify key risk-insights on operating LWR reactor technology.
- A Performance Monitoring Strategy to Enhance Consistency in Risk-Informed Decision-Making (ongoing) – Through review and evaluation of a sample of past reactor oversight process risk-informed decisions using a different approach focused on identifying sources of variability, this research effort is expected to identify potential enhancements to the existing self-assessment and feedback processes to help ensure consistency in decision outcomes (e.g., reduce bias, increase accuracy, and ensure transparency).
- Preparing Risk Assessment for Hydrogen Production and Use (ongoing) – This project seeks to update and improve the realism of PRA methodology that assesses the risks associated with hydrogen at NPPs for both on-site use and co-located hydrogen generation facility would be developed.

Collectively, these future focused projects help the agency build skills and knowledge to meet emerging challenges and opportunities in the continued implementation of risk-informed decisionmaking.

6. CONCLUSION

The NRC has continued to maintain a robust research portfolio focused on maintaining oversight of licensees, enhancing the use of risk-informed methods and approaches, preparing for new and advanced reactor deployment, and advancing the state-of-art and -practice in PRA technology. Through multiple partnerships with industry research organizations, national laboratories, other federal and international agencies, and universities, the NRC continually strives to address regulatory needs, enhance safety, and improve the efficiency of licensing and oversight processes.

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