Challenges for New and Advanced Reactor Licensing and Risk-Informed Applications: A Regulatory Perspective

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INTRODUCTION AND BACKGROUND

New and advanced reactor designs certified or licensed under the Code of Federal Regulations Title 10 Part 52 (10 CFR Part 52) are required to have probabilistic risk assessments (PRAs). Specifically, design certification (DC) applicants are required in 10 CFR 52.47(a)(27) to provide a description of the design-specific PRA and its results and combined license (COL) applicants are required in 10 CFR 52.79(a)(46) to provide a description of the plant-specific PRA and its results. Further, COL holders are required in 10 CFR 50.71(h)(1), by the scheduled date of initial fuel loading, to develop a Level I and II PRA that covers those initiating events and modes for which the Nuclear Regulatory Commission (NRC) has endorsed consensus PRA Standards that exist one year prior to initial fuel loading. COL holders are required in 10 CFR 50.71(h)(2) to maintain the PRA and upgrade the PRA every four years to cover initiating events and modes for which NRC-endorsed consensus PRA standards exist one year prior to the upgrade.

In addition to gaining insights from the PRA that supports further enhancing the designs and demonstrating NRC requirements and expectations are met, the PRAs are used to support the following programs:

- Regulatory Treatment of Non-Safety Systems (RTNSS)
- Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)
- Reliability Assurance Program (RAP)
- Future aspects of Regulatory Oversight, Technical Specifications, the Maintenance Rule (10 CFR 50.65), and others

To fully gain the benefits of developing and maintaining a PRA through the various design and licensing stages, new and advanced reactor applicants and licensees are also considering a variety of risk-informed applications. Depending on the stage of design or licensing there are unique challenges that arise in developing these applications and in the NRC review and approval of them.

From September 18, 2012 through February 19, 2013 the NRC held four public meetings to discuss risk-related topics for new reactors. These meetings included discussions on a number of risk-informed applications, including:

- Risk-Informed Inservice Inspection (RI-ISI)
- Risk-Informed Inservice Testing (RI-IST)
- Risk-Informed Categorization and Treatment (RI-C&T) per 10 CFR 50.69
- Maintenance Rule (MR) per 10 CFR 50.65
- Risk-Managed Technical Specifications (RMTS)
- Risk-Informed Surveillance Frequencies (RI-SF)

This paper provides a regulatory perspective on a number of the challenges that have been identified with risk-informed applications for new and advanced reactors and offers some possible strategies for addressing these challenges.

RISK-INFORMED APPLICATION CHALLENGES

Based on the staff interactions at the various public meetings, the issues encountered with risk-informed applications for new and advanced reactors can be separated into two basic groups: process issues and technical issues. Both of these types of issues are discussed below.

Process Issues

The process issues are typically associated with requirements for the specific risk-informed application that cannot be achieved at the particular design or licensing phase. The process issues are also related to the relative timeframe between development, submittal, and implementation of risk-informed applications.

A number of risk-informed applications require some type of expert panel or integrated decision-making panel (IDP) and identifies the minimum makeup of these panels that cannot be met at some design and licensing phases. For example, 10 CFR 50.69 establishes that the IDP must be staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering. The guidance in Nuclear Energy Institute (NEI) 00-04 "10 CFR 50.69 SSC Categorization Guideline," which is endorsed by the NRC in Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," further states: "Two key areas of expertise to be emphasized are experience at the specific plant being evaluated and experience with the plant-specific risk information relied upon in the categorization process." Even though such an application might be submitted during the COL applicant phase, the IDP members would not meet the qualification requirements and would not represent specific plant operational experience during this phase. As such, the ability to rely on the IDP to support the risk-informed application is limited until plant operational experience can be obtained.

Another issue is the ability to identify the appropriate functional attributes of what constitutes a technically acceptable PRA for the specific risk-informed application; considering what is the achievable scope, level of detail, and technical adequacy of the new or advanced reactor PRA at a particular stage (i.e., DC applicant, COL applicant, COL holder, initial operations). Many of the risk-informed applications establish the level of technical adequacy (e.g., Capability Category II) based on the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sb-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." For new and advanced reactors, many of the supporting requirements in the ASME/ANS PRA Standard will be difficult to demonstrate at this level until sufficient operating experience is gained. For example, achieving Capability Category II for many of the supporting requirements pertaining to data for the PRA requires that plant-specific data be used. For new and advanced reactors, there will be no plant-specific component performance data to achieve this capability category at the time of the application and for initial operations.

All risk-informed applications currently being pursued were originally developed for operating reactors that recognized the limitations in the scope of many plant-specific PRAs and provided alternative approaches for licensees to use under these circumstances (e.g., not having a fire PRA). However, COL holders will have, by the time of initial fuel load, relatively complete PRA models that address all hazards at full power. The NRC staff would expect that the alternative approaches would not be used by licensees that have a PRA for the specific hazard and mode.

Technical Issues

The technical issues are typically associated with the lack of operational information and experience. This lack of knowledge and experience impacts the ability to fully identify and address component failure modes, degradation mechanisms, potential cross-system common cause failures, and uncertainties. Further, there are some new reactor designs where new components are being used or components are being used in new ways, such as squib valves. The applicability of generic data under these new operating and environmental conditions further limits the ability to fully identify and understand component performance.

For some risk-informed applications there are questions regarding the appropriateness or applicability of the method or code that is relied upon in the existing guidance that may be acceptable for current operating reactors. For example, the ASME Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1," has been accepted for RI-ISI relief requests for some current operating reactors. In general, this approach replaces a detailed evaluation of the safety significance of each pipe segment with a generic population of high safety-significant segments based on experience during the previous decade and half of RI-ISI relief requests for current operating plants. The approach is supplemented with a plant-specific flooding analysis to confirm the selections and identify additional plant-specific high safety-significant segments. Such a reliance on current operating fleet experience may not be applicable to new and advanced reactor designs.

There are also technical issues associated with the scope of the PRA. For example, the proximity and integral design of several collocated small modular reactors may cause the need to consider the potential for multiple core damage events to occur nearly simultaneously due to a common event. For such designs, it may be necessary to develop new risk acceptance guidelines for licensing and risk-informed applications.

For applications such as 10 CFR 50.69 involving RI-C&T, it is likely that some functions for certain components will not be known until the plant is built and procedures are written and finalized. Further, the scope and level of detail of the PRA may change from COL application to initial operations as further design details are developed and modifications incorporated. The fact that the design, and hence the PRA, is evolving means that the final set of components in the PRA may not be finalized until the plant has finished construction and is preparing for initial operations.

Finally, there are issues in properly determining the technical adequacy of the PRA for the various design stages of new and advanced designs. An attempt at addressing technical adequacy for advanced light water reactor (ALWR) designs has been drafted by an

ASME/ANS PRA Standard working group. This draft appendix to the current ASME/ANS PRA Standard identifies what supporting requirements can be met and what supporting requirements need to be altered to address ALWR designs. However, challenges arise when trying to address the ability to meet these requirements at various design and licensing stages for the ALWR reactors. For example, regarding a supporting requirement for parameter estimation in the current ASME/ANS PRA Standard, a PRA using only generic information would be Capability Category I, while a PRA that calculates realistic estimates for the significant basic events would be Capability Category II, and a PRA that calculates realistic estimates for all basic events would be Capability Category III. The designation of Capability Category II or III at the design certification or licensing stage is not expected to be achieved and is acceptable for this supporting requirement during design certification and licensing, but a designation of Capability Category II or III may be required to be achieved prior to implementation of some risk-informed applications. Similar expectations should be applied when considering supporting requirements that address operating experience or procedures. Further, the expectations of an independent (or "peer") review, as defined in the ASME/ANS PRA Standard, would need to be addressed.

PATH FORWARD

To fully gain the benefit from the strengths and insights of developing and maintaining a PRA, the challenges to the development, review, and approval of risk-informed applications for new and advanced reactors need to be recognized and strategies need to be developed to address the challenges. These strategies may involve revising endorsed guidance and standards or developing new guidance and standards specific to new and advanced reactors for licensing and risk-informed applications that address the following:

- specific PRA model limitations associated with the different stages of design and licensing and/or
- implementation guidance and/or implementation inspection guidance to ensure that the risk-informed application actually implemented at an operating new or advanced reactor, including use of the plant-specific PRA, is consistent with the process as approved and reflects the as-built, as-operated plant.

CONCLUSION

The risk-informed applications currently being considered by new reactor DCs and COLs, and the associated PRA guidance, were originally developed for operating reactors that were, for the most part, already built and operating, and for which there was significant equipment performance data and operating experience. The lack of operational history is one of the unique challenges to the NRC's review and approval of these risk-informed applications. The NRC recognizes these challenges and is currently working on various strategies to address the challenges.

ACRONYMS

ADAMS	Agencywide Documents Access & Management System
ALWR	advanced light water reactor
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
COL	combined license
DC	design certification
FR	Federal Register
IDP	integrated decision-making panel
ITAAC	Inspection, Test, Analysis, and Acceptance Criteria
MR	Maintenance Rule (10 CFR 50.65)
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessments
RAP	Reliability Assurance Program
RG	Regulatory Guide
RI-C&T	Risk-Informed Categorization and Treatment (10 CFR 50.69)
RI-ISI	Risk-Informed Inservice Inspection
RI-IST	Risk-Informed Inservice Testing
RI-SF	Risk-Informed Surveillance Frequencies
RMTS	Risk-Managed Technical Specifications
RTNSS	Regulatory Treatment of Non-Safety Systems

REFERENCES

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