Probabilistic Risk Assessment Developments for the Licensing Implementation of Risk-Informed Programs for Light Water Reactors

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Abstract: Probabilistic risk assessments (PRAs) have been relied on to various degrees of importance in decision making by the U.S. Nuclear Regulatory Commission (NRC) to complement its deterministic approach for many years. The use of PRA in regulatory decision making and licensing activities for U.S light water reactors has increased in recent years due to licensees acting to adopt many risk-informed initiatives. Appropriately applied, risk assessments enable licensees to add flexibility to plant operations, and they enable licensees and the NRC to distinguish and focus on more safety significant from less safety significant issues. Lack of clarity in the use of risk assessments in the risk-informed integrated decision-making process caused some reviews to be dominated by deterministic considerations and evaluations, and created obstacles to fully examine all the anticipated benefits of using PRAs in the regulatory framework. Risk-informed regulations, regulatory guides, and guidance documents were crafted to balance the deterministic and probabilistic considerations to facilitate the significant growth of risk-informed activities, but challenges exist to implement a fully integrated evaluation of such initiatives.

This paper examines the latest NRC insights and challenges in the increased use of risk information in license amendment requests for existing nuclear power reactors regarding the implementation of these risk-informed programs.

Keywords: Operating Reactor Applications, Plant-Specific PRA, Regulations, Regulatory Guidance

1. INTRODUCTION

In the past few years, the application of probabilistic risk assessment (PRA) technology to enhance US nuclear power plants operational flexibility has increased dramatically. Transitioning away from a deterministic framework that has dominated the regulatory and operational framework has produced insightful questions regarding the quality of risk information and to what extent that information should be used in facility operation. Despite refocusing the PRA results for use in operations, the goal remains on ensuring that the facility remains safe using technically defensible information.

While there may be some debate to the genesis of the practical use of PRA, many agree on the effectiveness of the efforts that went into the development of, and the significant lessons learned from, the Individual Plant Examination (IPE) and its external events counterpart, the Individual Plant Examination for External Events (IPEEE), in expanding the use of PRAs. Following the NRC's August 8, 1985 policy statement on severe accidents [1], which concluded that existing plants pose no undue risk to the public health and safety, this systematic evaluation asked licensees to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission. Many of those insights served as precursors to bridge the use of risk in operations. However, there were limitations to meet its objectives, such as the lack of fully quantitative PRA models for some hazards, as well as level of detail or realism in PRA models.

Some years later, in its “Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” dated August 16, 1995 [2], the NRC encouraged the use of Probabilistic Safety Analysis (PSA, currently referred to as Probabilistic Risk Assessment or PRA) in nuclear regulatory activities. That Policy Statement stated that “…the use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC’s deterministic approach.” Having a PRA model that can quantify the risk from multiple hazards would support this objective. Following the PRA policy statement, use of risk analyses began to expand to address technical challenges in compliance, such as Code and Standards, and to support changes to licensing bases. In response to the PRA policy statement, in 1998 the NRC published Regulatory Guide (RG) 1.174 [3] to provide guidance on the use of PRA findings and risk insights in support of licensees’ requests for changes to a plant's licensing basis. At that time there was no standard for a PRA.

In parallel, starting in mid-1980s, the NRC started applying the results from PRA to create mandatory rules: the anticipated transients without scram (ATWS) rule in 10 CFR 50.62 [4], published in 1984, the station blackout (SBO) rule in 10 CFR 50.63 [5], published in 1988, and the maintenance rule in 10 CFR 50.65 [6], initially published in 1991 and amended in 1999. The ATWS rule has requirements for reduction of ATWS risk requiring each reactor to have equipment, independent and diverse from the reactor trip system, that would automatically initiate actions to mitigate the consequences of an ATWS in a reliable manner. The SBO rule contains requirements for coping with Station Blackout and was developed based on insights gained from several plant-specific probabilistic safety studies; operating experience; and reliability, accident sequence, and consequence analyses completed between 1975 and 1988. The objective of the rule was to reduce the risk of severe accidents resulting from SBO by maintaining highly reliable AC electric power systems and, as additional defense-in-depth, assuring that plants can cope with a Station Blackout for a specified duration. Finally, the maintenance rule in 10 CFR 50.65 requires reactor licensees to monitor the performance or condition of Structures, Systems and Components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. In an amendment to the rule in 1999 the NRC added the requirement that, before performing maintenance activities (including but not limited to surveillances, post maintenance testing, and corrective and preventive maintenance), the licensees assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to SSCs that a risk-informed evaluation process has shown to be significant to public health and safety. Although not explicitly required by the rule, the licensees have been using internal events PRA models to perform these risk assessments and continue to do so today.

Prior to the establishing of the consensus PRA Standards and the risk-informed licensing paradigm that involves a consistent level of PRA technical capability for licensing amendment applications, the nuclear industry recognized the need to develop a standard that would determine PRA quality. The Boiling Water Reactors Owners Group (BWROG), Pressurizer Water Reactors Owners Group (PWROG), and the Combustion Engineering Owners Group (CEOG), each generated processes and checklists that assessed the plant specific PRA to ensure it met a consistent level of quality. These efforts were improved upon and culminated into one of the first peer review process guidelines, Nuclear Energy Institute (NEI) 00-02 [7]. It provided guidance for the peer review of PRAs and subtier criteria for assigning a grade (i.e., Grade 1, 2, 3 or 4) to each PRA sub-element.

Just as the industry recognized the need to develop standards, the NRC saw the importance in establishing confidence in the information derived from a PRA. NRC staff needed a way to ensure the accuracy of the technical content is sufficient to justify the application specific results and insights used to support a regulatory decision. Discussions about the need for a PRA peer review process, as opposed to a detailed review by the NRC staff of the licensee’s PRA, started when the NRC began rulemaking on 10 CFR 50.69 [8] which gives licensees flexibility in providing special treatment to SSCs using risk insights.

The staff informed the Commission about the burdens imposed on licensees to establish and maintain PRA quality for risk informed initiatives such as 10 CFR 50.69 [9]. In a 2003 memorandum [10], the then Chairman of the NRC indicated that a policy decision was needed to stabilize the PRA quality expectations and requirements and enable its broader and more predictable use in safety related applications. In response, in 2004 the NRC staff developed a plan [11] for a phased approach to PRA quality and articulates how the staff would rely on the peer review process, the ASME/ANS (American Society of Mechanical Engineers/ American Nuclear Society) PRA standard, and RG 1.200 to establish and maintain PRA quality.

The first revision of RG 1.200 [12] was issued in 2004 and endorsed, with clarifications, the PRA standard and the NEI 00-02 peer review guidance provided by the standards-setting and nuclear industry organizations. Instead of an in-depth NRC review of the PRA during licensing actions, this regulatory guide allows the NRC staff to focus on the results of the peer review process, which is performed by experienced PRA practitioners against a specified set of requirements detailed in the PRA standard. In Revision 0 of RG 1.200, the NRC staff provided its position on NEI 00-02 that calls for a comparison, or self-assessment, to Capability Category II in the endorsed version of the 2002 ASME/ANS PRA Standard [13].

The ASME/ANS PRA Standard was endorsed by the NRC in RG 1.200 to be one acceptable approach for licensees to demonstrate their plant specific PRA is technically acceptable. It is a set of requirements on what a PRA must contain. It is important to note that the use of the word “requirement” is Standards language and the use of the word is not meant to imply a regulatory requirement. Over the years this PRA standard expanded from initially addressing internal events PRA, to addressing all hazards. The 2002 ASME/ANS Standard, as endorsed by RG 1.200 Revision 0, addressed full power, internal events (excluding fires) and limited Level 2 PRA. The 2005 ASME/ANS PRA [14], as endorsed by Revision 1 of RG 1.200 [15], applied lessons learned and insights to the internal events and Large Early Release Frequency (LERF), and introduced fire PRA supporting requirements. The 2009 ASME/ANS PRA standard [16], as endorsed by RG 1.200 Revision 2 [17], provides both process and technical requirements for an at-power Level 1 and limited Level 2 PRA for internal events, internal flood, internal fire, seismic, wind, external floods and other external events.

As the PRAs evolved and PRA Standards were refined over the years, the use of risk information broadened to allow the use of several different initiatives and purposes. Voluntary processes with expected benefits and flexibilities while maintaining adequate safety, facilitated widespread interest in the nuclear industry. Risk information expanded many regulated programs (e.g. Inservice Inspections, Extension of Containment Integrated Leakage Rate Testing Frequency, etc.). These programs and processes served a purpose for many years as the methods to ensure the plant and all its SSCs were being adequately maintained. Risk-informed initiatives improved the efficiency of these processes by assessing the change in the overall risk as the result of changes to operation and maintenance of SSCs. Among other considerations, assessment of change in risk allowed licensees to prioritize the most significant SSCs for inspection and maintenance. These processes were submitted to the NRC for years as a part of the license amendment request process until widespread adoption of today’s risk-informed programs occurred.

One major initiative that promoted the shift from primarily traditional engineering approaches was the Risk-informed Technical Specifications Initiative (RITS). Technical Specifications dictate plant operations by defining which SSCs must be in service, how long SSCs can be out of service before certain actions would be required, and required surveillance testing to establish whether an SSC is operational. This fundamental role to the operation of a commercial nuclear facility established the appropriate level of assurance for SSCs needed to prevent and, if necessary, mitigate accidents and transients. Risk information has permitted some level of flexibility with a few initiatives, while ensuring that safety is maintained. To improve efficiency, the Technical Specification Task Force (TSTF) submitted proposed generic changes to the NRC, Travelers, that encouraged adoption of processes through a template. Two notable risk-informed Technical Specifications initiatives were Initiative 4b, related to risk-informed completion times, and Initiative 5b, related to risk-informed surveillance frequencies.

Another major risk-informed initiative is the adoption of the National Fire Protection Association (NFPA) Standards Standard 805 [18] which was incorporated in 2004 in 10 CFR 50.48(c) [19]. NFPA 805 describes a methodology for existing light-water nuclear power plants to apply risk-informed, performance-based requirements and fundamental fire protection design elements to establish fire protection systems and features, as opposed to the deterministic fire protection requirements in 10 CFR 50.48(b) [19], otherwise known as Appendix R. The deterministic requirements were developed before the staff or the industry had the benefit of PRAs for fires and before recent advances in performance-based methods. The NFPA 805 risk-informed voluntary alternative reduced the need for exemptions and unnecessary regulatory burden associated with the deterministic fire approaches and adds flexibility to licensees' fire protection activities, while maintaining safety. This initiative was the main driver for development and improvements in the fire PRAs, which are also widely used for the Technical Specifications Risk Informed Completion Times program, Initiative 4b. NFPA 805 was adopted by 40 units of the 96 reactor units in the US.

The work in this paper was motivated by the need to contextualize the increased use of risk information in license amendment requests for nuclear power reactors. It will also present the benefits of expanding the use of PRA in a practical context. Finally, it will clarify challenges with using more PRA in a deterministic paradigm.

2. Guidance for PRA

The increased use of risk information in licensing actions served as a major contributor to the development of multiple guidance documents and regulations governing acceptable uses of the technology. One avenue was the Regulatory Guide series. Each Regulatory Guide provides guidance to licensees on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. Initially, licensing changes were characterized as changes that required explicit acceptance within established staff positions. There were also licensee-initiated changes to the current licensing basis which did not require NRC review and approval (e.g., changes to the facility as described in the Final Safety Analysis Report, which are the subject of 10 CFR 50.59 [20]).

RG 1.174 [21] is one such guide which applies the use of PRA in support of decisions to modify an individual plant's current licensing basis. This Regulatory Guide established an acceptable approach for developing risk-informed applications to support plant-specific licensing basis changes. To implement this type of decision-making in RG 1.174, the NRC staff expects all changes to meet a set of key principles. These principles articulate the basis used for traditional engineering and risk analysis decisions. The five key principles are shown in Figure 1.

These five key principles formulate the integrated decision-making construct that the NRC applies to all risk-informed changes to licensing basis. The hallmark to these principles is the first three principles which make up the traditional engineering analysis. Many changes to licensing basis documents today require an appropriate level of consideration and attention to ensuring these important principles are met. RG 1.174 provides detailed discussion on the level of detail and criteria considered. Additionally, key principles 4 and 5 use risk information that feeds into the decision on whether to accept the change.

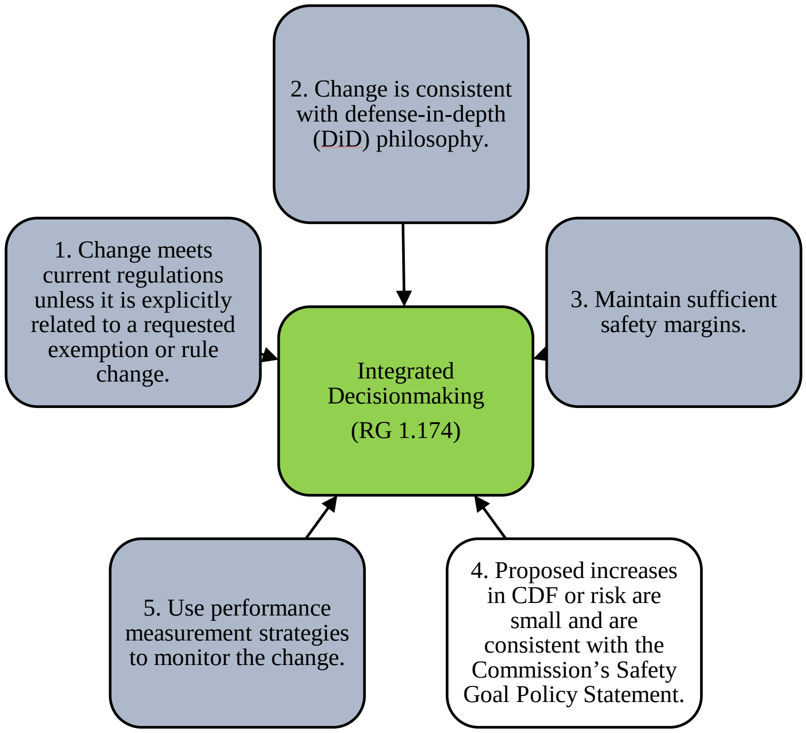

Figure 1 The five risk-informed principles of RG 1.174

For a PRA model to meet key principles 4 and 5, the NRC staff evaluates whether the PRA is of sufficient scope, level of detail, conformance to technical elements, and plant representation for the intended application. Results from the PRA are used to assess the overall impact of the risk and the change in risk due to the requested change. Licensees also characterize the impact of PRA uncertainties on the change in the licensing basis. The acceptability of the PRA is a multi-faceted process: one part which involves evaluating the base model against the criteria set forth in RG 1.200 and the other part which requires considering specific elements of the risk model against the plant-specific change being considered. Usually the level of review by the NRC staff of the PRA technical adequacy is directly related to the reliance on the PRA for the decision under consideration. Figure 2 shows for recent risk-informed applications how the NRC level of review varies depending on how much the PRA is relied upon in the regulatory decision.

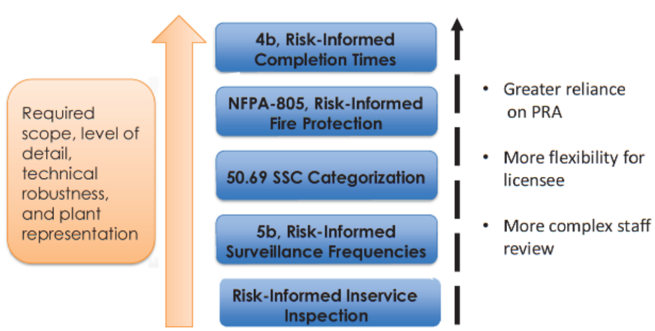

Figure 2 Level of PRA acceptability depending on the risk-informed application

As discussed above, RG 1.200 [17] serves as the guidance document to demonstrate the base PRA model is technically acceptable. Licensees submitting their PRA model for acceptability in licensing basis changes receive a peer review and ensure that the model has been compared to the endorsed ASME/ANS PRA Standard [16]. Licensees usually describe a combination of peer reviews and assessments, such as full scope (reviews of the full PRA model considered against all elements of a hazard in the standard); focused scope (reviews against particular elements within a hazard in the standard); self-assessments (assessments of PRA models peer reviewed to previously endorsed standards); and closure reviews (independent assessments to close-out peer reviewed findings to obviate need for detailed NRC review [22]).

Using the same principles of RG 1.174, RG 1.177 [23] applies the use of PRA in support of decisions to modify an individual plant's Technical Specifications. Prior to the development of RG 1.177, nuclear facilities considered PRA insights as a part of NRC's review of improvements to Technical Specifications. Most of the changes were extensions to allowed outage times or surveillance test intervals. The NRC staff notes that the improved Standard Technical Specifications [24-28] use the terminology "completion times" and "surveillance frequency" in place of "allowed outage times and surveillance test interval," respectively.

RG 1.177 describes several specific considerations that establish an acceptable risk-informed approach and additional acceptance guidance geared toward the assessment of proposed TS changes. As a part of key principle 4 (i.e. risk increase principle), RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed TS change, as discussed below:

- Tier 1 assesses the overall impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF), consistent with RG 1.174. It also evaluates increase in plant risk when equipment covered by the proposed Completion Time (CT) is out of service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses PRA the technical adequacy of the licensee's plant-specific PRA for the subject application. Finally, Tier 1 considers the cumulative risk of the present Technical Specifications (TS) change in light of past (related) applications or additional applications under review, along with uncertainty/sensitivity analyses with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved.

- Tier 3 addresses the licensee's overall configuration risk management program to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and that the licensee takes appropriate compensatory measures to avoid risk-significant configurations that may not have been considered during the Tier 2 evaluation.

In addition to the NRC staff evaluating the merits of the PRA for licensing-basis change, the staff also considers one-time TS changes.

In addition to guides discussed earlier, there are several other NRC guidance documents that provide more detailed guidance for specific risk-informed applications, such as RG 1.205 [29] for risk-informed performance-based fire protection program and RG 1.201 [30] for risk-informed categorization of SSCs, all of them being consistent with RG 1.174.

3. Insights and Challenges of Increasing the Use of Risk Information

Examining the increase in the use of risk technology in operation of nuclear facilities has assisted in the production of many observations of its expanded development and application. Many programs, such as the Risk-Informed Technical Specification Initiatives, have received much attention and research since the early 2000s. Other programs, such as risk-informed categorization and risk-informed fire protection programs, have found a way into plant operations via regulations.

As the flexibility in plant operations increased, there arose a need to ensure the PRA models are capable of quantifying risk commensurate with the flexibility that is approved. Unlike early risk analyses which emphasized on identifying vulnerabilities in plants and commonly employed conservative considerations, many PRA models today can extract discreet functional impact of specific components for more accurate analysis. The accurate analyses allow the programs that exist today to realize their full potential.

Developments in Risk-Informed Initiatives set up a framework that allowed PRA to be used in many aspects of operation, most notably Completion Times and SSC categorization. These broad scope amendments covered many SSCs and required input from multi-disciplinary teams to develop. While the ever-changing landscape of risk assessment will shape the risk insights today, let’s examine the benefits and challenges of the latest developments.

3.1. Risk-Informed Completion Times (RICT): Technical Specification Task Force (TSTF) - 505

TSTF-505 is the implementation guidance for Risk-Informed Technical Specifications Initiative 4b. It is a risk-informed process to temporarily extend certain Technical Specification completion time (CTs) up to 30 days, based on plant specific configuration and real-time risk calculation. Based on NEI 06-09 Revision 0-A [31], this process adds a new program in TS "Administrative Controls" entitled the "Risk-Informed Completion Time Program.” TSTF-505 Revision 2 [32] provides a list of Standard Technical Specification (STS) Conditions in scope, model license amendment application, and model safety evaluation. It also introduced the concept and application of PRA Functionality.

TSTF-505 and Initiative 4b reviews necessitate a large, multi-disciplinary review team. Early on, many challenges stemmed from a lack of understanding related to the use and application of PRA in plant operations. The process broadened the use of PRAs in the licensing basis and therefore touched many different systems and disciplines. The review of PRA acceptability for risk informed completion times submittals is the most complex review by the NRC and the level of PRA information required to support these applications is consistent with that understanding.

Many conditions within scope of TSTF-505 required some level of deterministic confirmation to ensure the plant remains capable of performing its design basis function when entering each of the configurations associated with a completion time. There were questions from traditional engineering groups at first but detailing the insights and assumptions from the PRA model gave those groups confidence that the SSCs and applicable functions matched their understanding of how the systems operated.

One of the most significant topics to come up during the early reviews of TSTF-505 was the use and application of PRA Functionality. Once a licensee finds itself in a Technical Specification Action Statement and is implementing a RICT, the licensee may find that the SSC has reduced capability but can still perform all or a few of its remaining functions. In this case, NEI 06-09-A, Section 2.3.1 allows the use of PRA Functionality to be credited in a RICT, provided it meets the criteria listed in that section. According to NEI 06-09-A, a component may be considered PRA functional for purposes of the RICT calculation if it was declared inoperable due to degraded performance parameters, but the affected parameter does not and will not impact the success criteria of the PRA model. The degraded condition must be identified and its associated impact to equipment functionality known and further additional degradation that could impact PRA functionality is not expected during the RICT.

The major benefit of PRA functionality is the opportunity to credit the SSC's actual capability. Technical Specifications cover a plethora of possible conditions but does not always achieve a discrete level in terms of measuring why an SSC is inoperable. The three PRA functionality criteria attempt to characterize the SSC inoperability against the success criteria of the PRA model. Let's examine a bit further the benefits and challenges to the PRA operational framework.

Measuring the actual capability of an SSC has been a feature of the Operability determination process for some time. Nuclear facilities measure the capability of components to perform its Specified Safety Function, which determines the Operability of SSCs. PRA Functionality, within the framework of the RICT program, would take the engineering understanding and apply success to the components where appropriate. Understanding the true capability of the SSCs allows for more accurate reflection of the as built, as operated plant.

Measuring the true capability of an SSC provides a great benefit as the impact of the outage for a particularly risk significant SSC could change several key factors in the facility. One of those factors is risk accumulation. While pre-planned maintenance can be set-up to limit this accumulation; emergent failures can challenge the amount of time available to be in a RICT as well as the required periodic reassessment of the overall risk impact of the program in terms of change in risk.

Another factor of measuring the capability is also unavailability. A completion time extension may increase the unavailability of an SSC due to the increased time the component is permitted to be out-of-service for maintenance or repair. However, an SSC whose parameters can meet its PRA success criteria would have reduced unavailability as compared to a normal outage without the RICT program. The benefit of such a feature has impact to other oversight program and applicability.

A major challenge to the use and application of PRA Functionality is the potential introduction of a new operating state. Until the development of the RICT program, TS defined two states for SSCs: Operable and Inoperable. While it may have been understood that there were varying level of functionality to Inoperable SSCs; there was no mechanism to standardize that "level of Inoperability" for a Completion Time. PRA success criteria brings together multiple parameters using analysis tools such as MAAP (Modular Accident Analysis Program). When using those tools to define whether an SSC is PRA functional, potential challenges may arise in relying on those SSCs in a degraded state for an extended period. With PRA functionality, degraded flowrates and temperature limits are especially challenging since these parameters would be required to be in a steady-state and not be impacted by any of the work to be performed.

Conclusions Regarding TSTF-505

Overall, the evolution of risk into its applicability to completion times is a testament to the efforts on both the nuclear industry and the NRC to challenge its understanding of plant operations. While PRA success criteria may never replace design basis analysis, it is important to note that continued data collection on the impact of this operating state is the key to further development in risk-informing plant licensing bases.

3.2. Configuration Risk Management Tool

Another example of risk-informing plant operation is the integration of configuration risk management (CRM) software tools (e.g. EOOS [Equipment Out of Service], Paragon, Phoenix, etc.) into the quantification and characterization of plant configurations. This tool started first as a tool to meet the maintenance rule requirements in 50.65 and continued to be extended for use in TSTF-505. In TSTF-505, the PRA model is reviewed and accepted as a part of the licensing process with the expectation that the model will translate into a CRM software tool.

CRM tools are generally “one-top” fault tree models overlaid with a user-friendly graphical user interface. Most components are tied to specific basic events within the fault tree which would generate configuration specific risk estimates. It also recommends actions to manage the risk and provide cutsets to validate the risk analysis. The tool applies different software techniques and adjustments to increase efficiency of the quantification. As a result, licensee can quantify a configuration on the order of minutes.

Translation to the CRM tool may necessitate certain enhancements to the model to improve speed of calculation. While not necessarily extensive, there has been noted instances of adjustments to the model within the CRM tool (e.g. failing systems higher in the fault tree where specific components were modeled individually, considering certain SSCs always failed, etc.). These refinements may produce conservative change-in-risk results while others could potentially mask contribution of certain failures. Another challenge stems from the use of the software tool itself and the potential challenges with managing the software.

Configuration risk management tools have a beneficial impact to the facility overall, in that there is an increased awareness of risk concepts in all levels of plant operation. Raising that awareness allows safety to the public to be upheld by all licensing staff, especially when they understand risk. Daily work at facilities is normally a balancing act of multiple outages and maintenance configurations. The CRM tool would manage extended outages when plants have TSTF-505, limiting the need to request Notices of Enforcement Discretion. Furthermore, this tool will allow licensees to better track the accumulation of risk across the implementation of its risk-informed Technical Specification programs.

Conclusions Regarding CRM Tool

The responsibility for ensuring the plant configuration and extended completion time is accurate falls on the licensee. Quite a bit of work has been put into making this tool as efficient as possible without losing its accuracy. However, there are still challenges the NRC will examine going forward to ensure the tools quantification is accurate.

3.3. 10 CFR 50.69 Risk Informed Categorization

The voluntary rule in 10 CFR 50.69 [8] is a risk-informed program which applies a systematic risk-informed process to identify safety-significant versus low safety significant equipment at a plant based on the importance of the equipment to protect the public health and safety. The 10 CFR 50.69 (50.69) Rule exempts safety-related equipment that is found to be of low safety significance from meeting eleven special treatment requirements, such as quality assurance, inservice testing, inservice inspection, environment qualification, etc. The rule, rather than specifying alternate treatment for the safety related equipment of low safety significance, it is a performance-based rule, allowing the licensees to decide the alternate treatment, but requires the licensees to ensure, with reasonable confidence, that this equipment remains capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life.

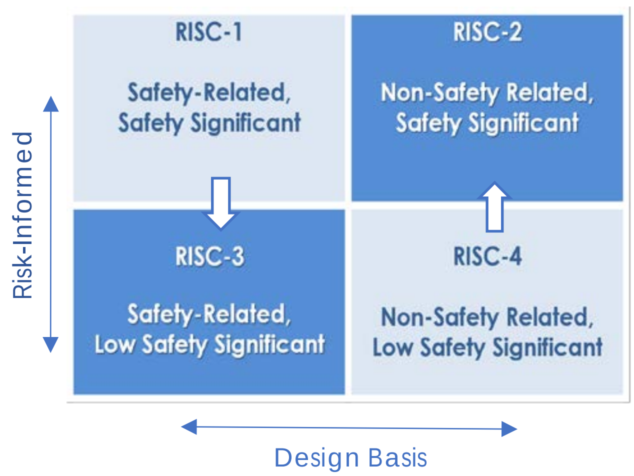

Figure 3 The risk informed safety classes defined by 10 CFR 50.69

While perceptions may infer this program as a risk-based program, it is a multi-disciplinary effort that applies risk with numerous deterministic "checks" along the way. The 50.69 process excels at balancing the input of different fields to find the appropriate considerations for equipment safety significance. Figure 4 shows an overview of the categorization process as described by the guidance in NEI 00-04 [33] as endorsed by RG 1.201 [30]. The core of the process is the Integrated Decision-making Panel (IDP) which is group of experts making the final decision on the categorization of SSCs.

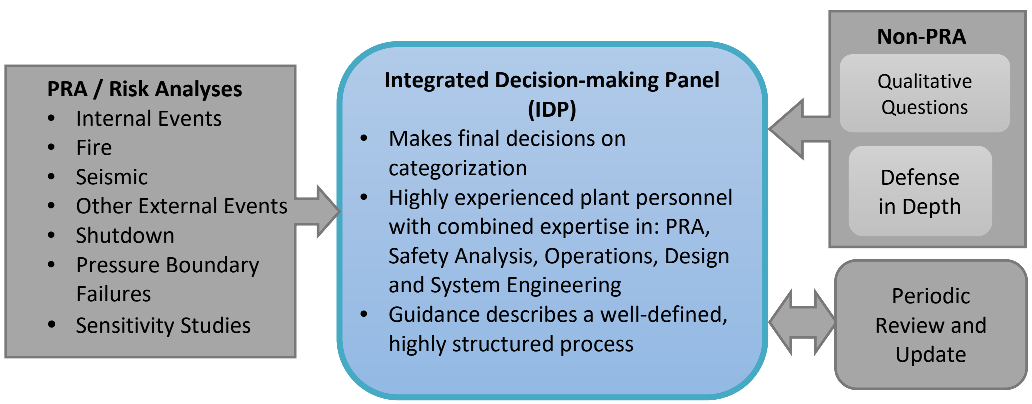

Figure 4 An overview of the 10 CFR 50.69 categorization process endorsed by RG 1.201

10 CFR 50.69 is frequently billed as a product that allows licensing facilities to achieve savings in testing, maintenance, and procurements programs. Using the PRA model as a part of this process allows the nuclear industry to inform its work surrounding safety-related and non-safety related SSCs.

The first challenge to this effort may be psychological. "Safety-related," equipment is the subject of many regulatory requirements, generally called special treatment, that are designed to ensure that all the equipment is of high quality and high reliability and have the capability to perform during postulated design basis conditions.

Secondly the effort for the licensees to prepare the License Amendment Request and support the regulatory licensing process through supporting audits and responses to requests for additional information is not trivial. First, the licensee has to demonstrate that it either implements the categorization process specified in the guidance in RG 1.201, or any proposed alternatives or deviations from the guidance are technically defensible and meet the rule requirements for an integrated, systematic categorization process. Secondly, the licensee must demonstrate that the PRA is technically adequate to support the program. This involves that the PRA has been peer reviewed, and any identified deficiencies have been addressed, or otherwise demonstrated that it does not impact the categorization process, meaning it does not significantly impact the PRA importance measures for SSCs and does not mask the contribution of certain equipment to the risk. It also involves a review of PRA assumptions and sources of uncertainties and disposition of any assumptions or sources of uncertainty determined to be key for the application. The licensee also has to propose a technical defensible way to address all hazards and plant operating modes, if PRA models for all hazards do not exist, which is typical for most plants. Once the license amendment request has been approved, the licensee is required to maintain their PRA to reflect the as-built and as-operated plant. Currently, future changes to the PRA does not require resubmission of a license amendment request. New license amendments requests are expected if the licensee is using a new PRA, such as a seismic PRA that has not been previously submitted in the 10 CFR 50.69 application to the NRC.

Third, the effort for the licensees to implement the program, perform the categorization, and implement alternative treatment is significant. The program necessitates multiple licensee organizations to be involved in the program, outside of the PRA practitioners, such as Engineering (Systems, Programs, Design, Procurement), Operations, Maintenance, Supply Chain etc. The licensee has to implement the procedures for categorization, prepare and deliver training to all plant staff involved prior to performing categorization. Per the rule, the licensee can select the systems to categorize, but once it selected a system, the entire system, and not parts of the system, must be categorized. Still, categorizing one system can be quite an involved effort, as systems contain in the range of few thousand components to tens of thousands of components. Each component must be systematically evaluated against all the aspects of the categorization process, including defense in depth assessment, PRA insights, shutdown safety, pressure-boundary considerations, other qualitative considerations, etc. Deciding on alternative treatment can also be a challenge and this is typically done at the industry level, not by the individual plants. The industry is still at the early stages of implementation of the program. There are few sites that categorized few systems, such as Radiation Monitoring System, Containment Spray, some cooling water systems, but the implementation of alternate treatment has been relatively limited.

Conclusions Regarding 10 CFR 50.69

Implementing 10 CFR 50.69, while it requires multiple deterministic checks, has transformed the way PRA is used. Categorizing SSCs and applying alternative treatment is yet another approach to allowing facilities to increase plant efficiency and maintain adequate safety.

3.4. Oversight Considerations

Looking into the future of incorporating PRA into plant programs such as those described above brings up the inevitable question of post-licensing implementation and regulatory oversight. Greater flexibility in licensing has necessitated a regulatory framework that can exist within the new paradigm. The NRC has performed many initiatives to examine its ability to provide adequate oversight and make the appropriate level of expertise available for support.

As knowledge develops further, the expertise associated with the inspection of the risk-informed licensing changes must evolve as the PRA model changes. The guidance for Risk-Informed Completion Times, NEI 06-09 Revision 0-A, requires regular model updates and maintenance, at least every other refueling outage. The rule in 10 CFR 50.69 has similar language. Licensees also make regular updates and upgrades to the PRA model as needed. The licensing basis information used for acceptance of license amendments is frequently supported by risk information which changes as knowledge increases.

A lot of the guidance regarding acceptability of PRA models has been established and is well understood. The review scope during license amendment requests is normally a sample of the peer review results. The peer reviews examine the PRA model, its methods, and documentation. Providing oversight in the PRA models presents some considerations on the scope and the role of peer reviews.

Oversight of PRA modeling in this new framework is effectively a continuation of past NRC efforts in the Reactor Oversight Process. The difference being that there has been an increased adoption of flexible operations programs. The transition of PRA models from licensing means there will be a greater interest in ensuring PRA models are documented in the appropriate amount of detail to support oversight post-licensing. The peer review process will still serve as an integral piece to establishing technically acceptable PRA models.

4. CONCLUSION

The risk-informed regulatory framework has evolved along with industry risk assessment developments. Transforming from a predominantly deterministic paradigm to a risk-informed approach has brought about some challenges, but also presented some useful insights at nuclear facilities. Further flexibility and development in risk-informed plant operations will serve to continue advancing the nuclear industry to increasing efficiency and effectiveness, while maintaining safety.

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