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MEMORANDUM TO: Russell Felts, Director
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FROM: Antonios Zoulis, Chief
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Signed by Zoulis, Antonios
on 05/24/23

SUBJECT: SUMMARY OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF OBSERVATIONS FROM THE PROBABILISTIC RISK ASSESSMENT CONFIGURATION CONTROL TABLETOP SITE VISITS

The purpose of this memorandum is to document the results from the Working Group's eight tabletop visits to different facilities with approved risk-informed programs. These tabletops were not inspections, but rather a voluntary effort, supported by NEI and industry, to facilitate the U.S. Nuclear Regulatory Commission staff's understanding of the implementation of the Probabilistic Risk Assessment Configuration Control programs and to assist in optimizing the NRC's future inspection guidance in this area. The Working Group was formed in January 2021 and included representatives from the Division of Risk Assessment and the Division of Reactor Oversight in the Office of Nuclear Reactor Regulation, as well as representatives from all four regions.

Enclosures:
As stated

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SUBJECT: SUMMARY OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF OBSERVATIONS FROM THE PROBABILISTIC RISK ASSESSMENT CONFIGURATION CONTROL TABLETOP SITE VISITS DATED: 6/21/2023

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PROBABILISTIC RISK ASSESSMENT CONFIGURATION CONTROL INITIATIVE

Summary of the U.S. Nuclear Regulatory Commission Staff Observations from the Probabilistic Risk Assessment Configuration Control Tabletop Site Visits

JUNE 2023

Enclosure

EXECUTIVE SUMMARY

The Office of Nuclear Reactor Regulation (NRR) established a Working Group to identify a balanced approach to provide some level of oversight/inspection guidance for the implementation of Probabilistic Risk Assessment (PRA) Configuration Control (PCC) programs for approved hazard group models, supporting risk-informed decisionmaking for programs, such as, Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69, National Fire Protection Association (NFPA) 805, Risk-Informed Completion Times (RICT), and Surveillance Frequency Control Program (SFCP). The Working Group plans to leverage a graded approach in developing the inspection guidance of PCC program implementation for approved hazard group models, which aims to ensure that the approved hazard group models technically reflect the as-built, as-operated plant.

As part of the development process, the Working Group conducted eight tabletop visits to different facilities with approved risk-informed programs. These tabletops were not inspections, but rather a voluntary effort, coordinated via NEI and industry, to help the U.S. Nuclear Regulatory Commission (NRC) staff understand the licensee's implementation of PCC programs and to assist in optimizing future inspection guidance in this area. The teams focused on the implementation of PCC rather than PRA Acceptability/Technical Adequacy or re-certification of hazard group models.

Generally, the staff found that the licensees are implementing their PCC programs in a sufficient manner to support risk-informed programs. However, it was noted that these tabletops helped the staff to identify potential vulnerabilities in the areas of documentation and over reliance on knowledge-based programs. Specifically, the staff observed that for most facilities, the engineering input monitoring process is well-defined. However, the process for monitoring operations, maintenance, component performance monitoring, and industry-wide operational experience seemed informal at some of the facilities. This informal approach relies on the skill of the PRA engineers, staff relationships, and meetings with industry owner's groups to raise awareness of issues for the PRA staff to evaluate. In the view of the staff, this led to some examples where failure events were screened from parameter updates with insufficient justification, potential failure modes were not modeled, and the lack of licensee evaluation for some industry-wide operating experience. The staff also saw some instances of parameter data not being updated in a timely manner.

The results of the tabletop visits provided valuable insights into the implementation of PCC programs and will facilitate the staff's efforts to optimize future inspection guidance. Overall, the staff determined that inspection guidance for PCC program implementation should be designed to ensure that the level-of-effort is commensurate with the risk-informed programs implemented at the licensee's facility. Additionally, the staff recommends that future efforts focus on the independent verification of the input monitoring and information collection and the PRA maintenance and upgrade aspects of the PCC programs per Section 1-5 of the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard, with the goal of ensuring PCC does not adversely influence key insights associated with risk-informed applications.

BACKGROUND

In 1995, the Commission issued a Probabilistic Risk PRA policy titled, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Affairs; Final Policy Statement,” 60 FR 42622. This policy statement encourages the use of PRA in all regulatory matters and states 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.” Section II of the policy, “Summary of Public Comments and NRC Responses,” states that the Commission will require PRA quality commensurate with the proposed application.

Risk-informed programs as a result of the Commission’s PRA Policy, require PCC to maintain approved hazard group models technically adequate, reflecting the as-built, as-operated plant. The path to PCC regulatory requirements depends on the risk-informed program. Risk-informed programs include:

- 10 CFR Section 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.”

- 10 CFR 50.48, “Fire Protection,” subsection (c), “National Fire Protection Association Standard NFPA 805,” or NFPA 805.

- Technical Specifications Task Force (TSTF) – 505, “Provide Risk-Informed Extended Completion Times,” Revision 2, or the RICT program.

- TSTF- 425, “Relocate Surveillance Frequencies to Licensee Control,” Revision 3, or the SFCP.

In January 2021, NRR, Division of Risk Assessment, PRA Oversight Branch established a Working Group (WG) to address an identified oversight gap in the Reactor Oversight Process (ROP) for the inspection of the implementation of the PCC programs. The main goal of the WG was to close the identified oversight gap by identifying a balanced approach to provide for some level of oversight and inspection guidance to monitor the implementation of PCC programs for approved hazard group models, supporting risk-informed decision making for the following programs: 50.69, NFPA 805, RICT and SFCP. To achieve this goal, the following objectives were executed:

- Develop a strategy of a balanced approach for the oversight of PCC, including a plan to conduct tabletops to gather observations regarding the licensee’s implementation of the PRA CC programs.

- In taking a balanced approach, develop preliminary thoughts and guidance for the inspection of the licensee’s implementation of their PCC programs for PRA internal and external hazard groups. Identify recommendations for the incorporation of such guidance into the ROP baseline inspections per inspection manual chapter 2515, “Risk-Informed Baseline Inspection Program,” and inspection level of effort.

- Conduct outreach with Industry stakeholders and the public to obtain feedback on the staff’s initiatives.

- Implement the proposed guidance at licensed facilities to optimize and gather information, observations and onsite feedback via the tabletops.

- Based on the observations gleaned from the tabletops, refine the recommendations for a balanced approach for the oversight of the licensee's implementation of their PCC programs.

- Develop training for the NRC inspectors on the oversight of PCC implementation.

- Develop an interim significance determination process framework for PCC issues.

The WG used a performance-based approach to develop the inspection guidance of the PCC program implementation for approved hazard group models. Future inspection efforts will focus on the independent verification for the implementation of the PCC programs per Section 1-5 of the ASME/ANS PRA Standard.

- Engagement with the public and industry stakeholders commenced on February 2, 2022, with the first of three public workshops on PCC. During this meeting, the staff introduced the PCC framework as well as other relevant topics associated with this initiative. (The meeting summary is available in Agencywide Documents Access and Management System (ADAMS) Accession No. ML22066B274). On April 5, 2022, the NRC conducted a second PCC workshop, where the staff provided details of the tabletops to support the PCC framework at ML22129A175.

In coordination with industry stakeholders, eight different licensees agreed to host PCC tabletops to inform the PCC framework, as well as the WG's decisions and recommendations of a balanced approach for oversight. This opportunity provided the WG with a representation of the current PCC programs and the licensee's implementation of the programs. The eight facilities visited by the staff have the below combinations of approved risk-informed programs:

- Facility A: SFCP & 50.69
- Facility B: SFCP, RICT, 50.69 & NFPA 805
- Facility C: SFCP, RICT & NFPA 805
- Facility D: SFCP, RICT & 50.69
- Facility E: SFCP & NFPA 805
- Facility F: SFCP & NFPA 805
- Facility G: SFCP & RICT
- Facility H: SFCP, RICT & 50.69

The staff conducted the third workshop on March 30, 2023, where the NRC presented the proposed PCC regulatory path forward. The industry presented some lessons-learned that they gleaned from the tabletop evaluations, as well as ongoing and future industry actions. A summary of the meeting can be found at ML23115A301.

SUMMARY OF THE PCC TABLETOP SITE VISITS

The tabletop teams consisted of four NRC staff members with a mix of NRR and the Office of Nuclear Regulatory Research reliability and risk analysts, and regional and NRR senior reactor analysts. In general, the tabletops lasted one week, some performed at corporate offices where the PRA staff are located, that included a visit to the facility or at the facility for the entire duration.

These eight tabletops were not inspections, they were a voluntary effort, closely coordinated with NEI and industry to facilitate the NRC staff's understanding of the licensee's implementation of the PCC programs and to optimize the staff's future inspection guidance. The team's focus was on PCC rather than PRA Acceptability/Technical Adequacy or re-certification of hazard group models. Specifically, the NRC staff focused on plant changes to prevent duplication of the Peer Review process. Any potential issues identified by the NRC staff were communicated and provided as observations for the licensees to evaluate and take appropriate corrective actions, as necessary.

Overall, based on the reviews conducted to date, the NRC staff have confirmed that licensees are meeting the consensus standard; however, identified several observations (at some of the facilities) on how licensees are implementing their programs. Specifically, the NRC staff identified vulnerabilities in the areas of documentation and overreliance on knowledge-based programs.

The staff observed that all the PCC procedures for the eight facilities meet the intent of Section 1-5, "PRA Configuration Control," of the PRA ASME Standard. However, it was not apparent to the teams how the execution of input monitoring of some aspects, besides engineering, were accomplished during an update cycle, because the procedures did not provide further guidance for implementation. From Regulatory Guide (RG) 1.200, Revisions 2 and 3, the following inputs are to be monitored:

- Design
- Operations
- Maintenance
- Engineering

The teams observed that for most facilities, the engineering input monitoring process is well-defined. The engineering change (EC) process, via checklists, triggers the involvement of PRA engineers through the review process for changes impacting the internal events hazard group, and internal fire, for licensees with those hazard groups approved. For some of the checklists, it was not clear of a direct link between the changes impacting the internal flooding hazard group and when to involve the PRA engineers. For the monitoring of operations, maintenance and industry-wide operational experience, the process seemed informal at most of the facilities, relaying on skill-of-the-craft of PRA engineers, staff relationships, and meetings with industry owner's groups to raise awareness of potential industry experience for the PRA staff to evaluate. Internal site procedures in the areas of maintenance, operations, and industry-wide operational history, did not show a direct link for a formal trigger to involve the PRA organization, resulting from an input change that could impact the licensee's approved hazard groups (i.e., internal events, and/or internal flooding, and/or internal fire) to support risk-informed decision-making via approved risk-informed programs.

In the area of PRA maintenance and upgrade, the NRC staff observed that some facilities have formal databases to track maintenance log items, and some use the corrective action program database to track items needing further evaluation. The teams observed a general trend in documentation to determine if an input change warrants PRA maintenance, or an upgrade, or if the change could be screened out. Specifically, aside from responses to peer review documented findings, it was not apparent that facilities would verify if a supporting requirement

(SR) was impacted by the input change being dispositioned for maintenance, or upgrade or screen out.

NRC Staff Observations from the Tabletops

Provided below are the NRC staff's observations from some of the tabletops conducted.

Observation A:

The team reviewed a maintenance log item, which documented the vulnerability of both the motor and the turbine trains of Auxiliary Feedwater (AFW) to a major flood of the turbine building because of a plant modification to the Turbine Driven AFW (TDAFW) pump room. Specifically, the log item included a discussion of the vulnerability of the motor driven AFW (MDAFW) pump room via the failure to close the float check valves installed in the room drains, which are not part of the flooding hazard group model because it was not considered significant. The log item was closed to another log item, which documented the TDAFW pump room condensate drain line modification, which made the condenser pit common to both the TDAFW pump casings and the AFW motor driven pump room. The licensee concluded that the vulnerability of the failure of the float check valves in the MDAFW pump room was not modeled because 1) the probability of a check valve failure along with the frequency of the impacting flood is low; 2) the flood propagation analysis does not typically model a conditional failure probability of a failure of a barrier. The licensee entered this issue into its corrective action program for further evaluation.

The team reviewed this log item because of the high-risk significance of the AFW system determined by the Fussell-Vesely (FV) importance measure. This is an example the reliability of components that protect from a major flood were potentially not appropriately reflected in the internal flooding hazard group model, following a plant modification. This is an example of the internal events hazard model potentially not representing the as-built, as-operated plant.

Observation B:

The team reviewed an EC which installed two non-safety related diesel generators (DGs) capable of supplying a safety-related 4kV bus to either unit, for this dual unit site. Both engines provide up to 25% more power than a single safety-related emergency diesel generator (EDG). The DGs are synchronized into a 4kV (Bus X) bus through a programmable logic controller (PLC) that requires both engines to be operating and their individual output breaker closed before their tiebreaker is closed to Bus X. The system notebook for the DGs established a success criterion for one DG supplying 4kV power via Bus X to one 4kV safeguard bus at units 1 and or 2. For one DG to operate, emergency operating procedures (EOPs) direct operators to override the PLC and take additional steps to restore power available to a safety train in the station blackout (SBO) unit. The NRC staff reviewed the DGs representation in the PRA model and found that no Human Failure Event is included for PLC override as required by EOPs to accomplish single standby diesel generator (SDG) operations or a basic event for the failure of the PLC. The licensee entered this issue into its corrective action program for further evaluation.

This modification was chosen because of the high risk-significance of the DGs, as determined by the FV importance measure. The team reviewed failure events data for the DGs reported in the internal events data collection and analysis notebook, dated June 2020. The team noted that zero failures were reported for the DGs. The team compared the reported failures in the notebook to the Institute of Nuclear Power Operations Industry Reporting Information System (IRIS) database maintenance rule functional failures (MRFF) reported by the facility. The IRIS database showed one MRFF for the DGs within the notebook data collection period. The team questioned this discrepancy. The licensee explained that for non-MSPI components, PRA engineers submit a survey to the system engineers to collect failure event data for updates. The licensee entered this issue into its corrective action program for further evaluation.

The team expanded the review to the safety-related EDGs, which the notebook reported two failures. For the data collection period, six MRFFs were documented. The licensee entered this issue into its corrective action program for further evaluation.

The team reviewed this failure event data because of the high-risk significance of the DG's system determined by the FV importance measure and the EDGs, as an extent of condition. This is an example of data screened out for PRA parameter estimation without a documented justification.

Observation C:

The team reviewed an EC that involved a TDAFW pump suction header check valve replacement. The AFW system notebook established a success criterion for station blackout (SBO) scenarios requiring 375 gpm of auxiliary feedwater (AFW) flow to two of four Steam Generators for the first 4 hrs. of the event. This flow is to be maintained for at least another 2 hrs. for a total of 6 hours. The alternate water source from essential service water is connected to the suction of the TDAFW pump, upstream of the suction check valve. The team noted that the notebook assumed that the flow diversion from the supply lines from the MDAFW pump to the supply lines from the TDAFW pump, given failure of the TDAFW pump and vice-versa was not modeled. The team confirmed by reviewing the AFW fault tree that the failure to close of the suction check is not modeled. The AFW system notebook did not include a documented basis to address screening this out-flow diversion. The licensee entered this issue into its corrective action program for further evaluation.

The team reviewed this EC because the component was identified as critical. The licensee defines critical components as those with one or more failure modes, determined to be risk-significant in the internal events hazard group model. The licensee did not review this EC for impact on the internal events model because the replacement was like-for-like per the PCC process. This is an example of a general assumption for the internal events model without adequate documented justification to screen out a failure event.

Observation D:

The team requested a PCC evaluation of the Open Phase Condition (OPC) design vulnerability in the electric power systems as an input to industry-wide operational history. OPC as communicated to industry in NRC Bulletin 2012-01, "Design Vulnerability in Electric Power Systems," ADAMS Accession No. ML12074A115, was ultimately resolved by industry with Nuclear Energy Institute (NEI) 19-02, Guidance for Assessing OPC Implementation Using Risk Insights," ADAMS Accession No. ML19172A086, Revision 0. The executive summary of NEI 19-02 states, "This report provides guidance for the performance of a risk assessment to inform the decision of whether to implement the Open Phase Isolation System (OPIS) automatic trip function or to implement the OPIS to provide alarm indication to the control room operator and rely on proper operator action to diagnose and respond to the presence of an Open Phase Condition (OPC)." Following the guidance of NEI 19-02 does not relieve licensees of their PCC requirement to evaluate OPC as an industry-wide operational history input for potential maintenance or update for approved hazard groups in support of approved risk-informed programs. The licensee stated that it had not performed an OPC PCC evaluation. The licensee entered this issue into its corrective action program for further evaluation.

The team requested this evaluation due to the generic nature of OPC and to ensure that licensees are performing PCC evaluations because of this important operating experience. PCC ensures approved hazard groups reflect the as-built, as-operated plant due to input changes. OPC involves an input change resulting from industry-wide operational experience, highlighting an issue, that may require further systematic evaluations and potential engineering and operations changes to address, which could require a model maintenance or update. This is an example of the internal events hazard model potentially not representing the as-built, as-operated plant.

<u>Observation E</u>:

The team requested the latest data update performed by the licensee for review. The team found that the latest data update was performed in 2016, using performance data from January 2010 to January 2016, and generic data from the 2010 NUREG/CR-6928 update. The licensee justified the data update delay based on resources to implement RICT and 50.69 risk-informed programs with a qualitative evaluation, concluding that data updates typically do not have a large impact on the model. 10 CFR 50.69(e)(1) requires PRA updates shall be performed no longer than two refueling outages.

The team requested the latest data update because of the importance of updating approved hazard group models as the vehicle to implement performance measurement strategies required by RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," principle 5, "Use of performance measurement strategies to monitor change," since the development of the model. RG 1.200, Revision 2, PCC, and data analysis technical element requirements ensures this principle is met. The ASME PRA Standard, as endorsed by RG 1.200, Revision 2, provides the necessary tools to ensure this principle is met as well.

The team also requested the OPC PCC evaluation for review. The PCC evaluation concluded that the change in Core Damage Frequency to the Model of Record was small. However, an evaluation of the potential impact to the RICT calculation was not conducted. The licensee entered this issue into its corrective action program for further evaluation.

The team requested this evaluation due to the generic nature of OPC and because facility plant procedures allow a single offsite AC power source to feed multiple trains of equipment. Therefore, the facility would be sensitive to OPC events and represent a source of uncertainty in RICT calculations for facility power systems. This is an example of the internal events hazard model not potentially representing the as-built, as-operated plant.

<u>CONCLUSION</u>

The results of the tabletop visits provided valuable insights into the licensee's implementation of the PCC programs and will facilitate the staff's efforts to optimize future inspection guidance. Overall, the staff determined that inspection guidance for PCC program implementation should be designed to ensure that the level-of-effort is commensurate with the risk-informed programs implemented at the licensee facility. Additionally, the staff recommends that future efforts focus on a balanced approach for the independent verification of the implementation of the PCC programs per Section 1-5 of the ASME/ANS PRA Standard, with the goal of ensuring that PCC does not adversely influence key insights associated with risk-informed applications.

Appendix A: PRA Configuration Control Regulatory Background

In 1995, the Commission issued a PRA policy titled, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Affairs; Final Policy Statement,” 60 FR 42622. This policy statement encourages the use of PRA in all regulatory matters and states 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.” Section II of the policy, “Summary of Public Comments and NRC Responses,” states that the Commission will require PRA quality commensurate with the proposed application.

PRA Quality was first described in SECY-00-0162, “Addressing PRA Quality in Risk-Informed Activities,” (ADAMS Accession No. ML003732744) by establishing the scope and technical attributes of a PRA, as two areas for an appropriate level of confidence in PRA results for regulatory decision making. This description was interpreted in different ways by stakeholders, resulting in confusion and misunderstanding. SECY-04-0118, “Plan for Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality,” (ADAMS Accession No. ML041530055) defined PRA Quality as in RG 1.174 and RG 1.200 as having three aspects: the scope, level of detail and technical adequacy of the model. SECY-04-0118 stated, “Inherent in this definition is that a PRA of sufficient quality to support an application need only have the scope and level of detail sufficient to support that application, but it must always be technically adequate.” SECY-04-0118 presented RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 0, (ADAMS Accession No. ML040630078) as a trial use document to provide the level of confidence for PRAs technical adequacy by focusing the licensing reviews on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.

RG 1.200, Revision 0, Regulatory Position (RP) C.1, “Functional Requirements of a Technically Acceptable PRA;” described one acceptable approach for defining the technical adequacy for an acceptable PRA. RP C.1 provided guidance in three areas:

- The definition of the scope of a PRA
- The elements of a PRA
- The technical attributes and characteristics for a full-scope PRA

RP C.2, “Consensus PRA Standards and Industry PRA Programs,” presented one acceptable approach to meet RP C.1 using an industry consensus PRA standard or with the use of an industry developed peer review process as an alternative approach to the industry PRA standard. RP C.2 included Table 4, “Principles and Objectives of a Standard.” Within Table 4 the maintenance, and upgrades of PRAs to represent the as-built and as-operated plant was included as item 6. It also included Table 5, “Summary of Characteristics and Attributes of a Peer Review.” Within Table 5, reviews of PRA maintenance and update process was included. The RG endorsed, with exceptions (i.e., clarifications and qualifications), the ASME RA-S-2002, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” and Addenda A, ASME-RA-Sa-2003 as a consensus PRA industry standard that meets the guidance in RP C.2 for Level I PRAs. NEI 00-02, “Probabilistic Risk Assessment Peer Review Process Guidance,” Revision A3, was the endorsed industry standard for the peer-review process established in RP C.2.

RG 1.200, Revision 1, (ADAMS Accession No. ML070240001), RP C.1, "A Technically Acceptable PRA," added development, maintenance, and upgrade of a PRA as the fourth area for a technically acceptable PRA. The RG endorsed with exceptions to ASME RA-S-2002 and addendums A and B to the standard, ASME RA-Sa-2003 and ASME-Sb-2005. NEI 00-02, Revision 1, was the endorsed industry standard for the peer review process.

RG 1.200, Revision 2, (ADAMS Accession No. ML090410014), RP C.1 edited the four areas of a technically acceptable PRA covered to:

- Scope of a PRA
- Technical elements of a full scope Level 1 and Level 2 PRA and their associated attributes and characteristics
- Level of detail of a PRA
- Development, maintenance, and upgrade of a PRA

RP C.2 guidance for demonstrating compliance with RP C.1 changed from using the peer review process as an alternate to the consensus PRA standard to the current philosophy which requires an industry peer review to ensure the requirements from the consensus standard are met. NEI 00-02, Revision 1, NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, and NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 0 were endorsed for peer reviews. The RG discussed the NEI 00-02 "Licensee Self-Assessment Guidance" to resolve the differences between a prior version of the internal events standard (ASME Ra-Sb-2005), as endorsed in Revision 1 of RG 1.200, and its peer review criteria. The RG stated that the of self-assessments are to be used to demonstrate the technical adequacy of a PRA for an application, differences between the current version of the standard as endorsed in Appendix A and the earlier version be identified and addressed.

In 2007, the NRC issued Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," (ADAMS Accession Number ML070650428). As a result of this RIS, from 2010 and forward, risk-informed licensing applications have been submitted in accordance with (IAW) RG 1.200, Revision 2, and ASME/ANS RA-Sa-2009, as endorsed by the NRC, unless the licensee has incorporated a newer revision of RG 1.200 to maintain PRA Acceptability of risk-informed applications.

RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, (ADAMS Accession Number ML20238B871), introduced the term PRA Acceptability which had been synonymous with previously used terms such as PRA Quality and PRA Technical Adequacy[1]. RG 1.200 Revision 3 defines PRA Acceptability with respect to scope, the level of detail, conformance with the PRA technical elements (i.e., technical adequacy) and plant representation of a PRA position C.1.2, and how closely the PRA represents a plant's actual configuration and operations. Both RP C.1 and C.2 were re-named as "An Acceptable Base Probabilistic Risk Assessment," and "National Consensus Standards and Industry Programs for Probabilistic Risk Assessment," respectively. RP C.1 four areas were re-named as:

- Scope of a base PRA,
- Technical elements of a base PRA,
- Level of detail of a base PRA,

[1] See ADAMS Accession No. ML18024A766 for more information related to the term PRA acceptability.

- Plant representation and PRA PCC.

RP C.2.2 added guidance for peer review of upgrades or any newly developed methods (NDM). The RG, within Appendix B, endorsed ASME/ANS RA-Sa-2009, and ASME RA-S-Case 1, “Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications,” with exceptions. RP C2.2.4 endorsed industry guidance NEI 17-07, “Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard,” Revision 2, (ADAMS Accession Number ML19231A182), in its entirety as a means of satisfying the peer review requirements for the ASME/ANS RA-Sa-2009 PRA standard.

Risk-informed initiatives require PRA PCCs to maintain approved hazard group models as technically adequate, reflecting the as-built, as-operated plant. The path to PCC regulatory requirements depends on the risk-informed program. Risk-informed programs include:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,”

- 10 CFR 50.48, “Fire Protection,” subsection (c), “National Fire Protection Association Standard NFPA 805,” or NFPA 805.

- TSTF – 505, “Provide Risk-Informed Extended Completion Times,” Revision 2, or the RICT program.

- TSTF 425, “Relocate Surveillance Frequencies to Licensee Control,” Revision 3, or the SFCP.

Approved risk-informed categorization and treatment programs IAW 10 CFR 50.69 (50.69) include the PCC requirements per 10 CFR 50.69(e), “Feedback and process adjustment,” subsection (1), “RISC-1, RISC-2, RISC-3, RISC-4 SSCs,” requires licensees to review changes to the plant, operational practices, applicable plant, and industry operational experience and to update the PRA as appropriate. These reviews shall be performed in a timely matter but no longer than once every two refueling outages. 10 CFR 50.69(e)(2), “RISC-1 and RISC-2 SSCs,” requires performance monitoring of RISC 1 and 2 components for potential adjustments to categorization or treatment processes, as necessary. 10 CFR 50.69(3), “RISC-3 SSCs,” requires performance monitoring of RISC-3 components for potential adjustments to the categorization and treatment process.

Approved risk-informed fire protection programs IAW NFPA 805 per 10 CFR 50.48(c), modify their fire protection program license condition to include “Risk-Informed Changes that May Be Made Without Prior NRC Approval,” allowing licensees to change the program using risk assessments that are based on the as-built, as-operated, and maintained plant; and reflect the operating experience of the plant. In addition, NFPA 805 section 2.2.9, “Plant Change Evaluation,” directed the performance of a risk-informed plant change evaluation per section 2.4.4 for changes to previously approved fire protection program elements. Section 2.4.3, “Fire Risk Evaluations,” required PRA approach, methods, tools and data, used for performance-based evaluations of fire protection features and fire risk evaluations for change analysis described in section 2.4.4 to be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated, and shall be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant. RG 1.200, RP C.1 and C.2 are an acceptable way to demonstrate PRA Technical Adequacy per Revision 2 or PRA Acceptability per Revision 3.

Approved RICT programs are included in the Administrative Controls section of the Technical Specifications (TS) required the program to be implemented IAW NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." NEI 06-09-A, Rev 0, section 2.3.4, "PRA Technical Adequacy," item 2, required the PRA to be reviewed to the guidance of RG 1.200, Rev 0, for a PRA which meets Capability Category (CC) 2 for the SRs of the ASME PRA Standard. It also required deviations from CC 2 to be justified and documented. Section 2.3.4, "PRA Technical Adequacy," item 7, required the PRA to be maintained and updated in accordance with approved procedures to ensure it accurately reflects the as-built, as-operated plant. The maintenance and update process should include: 1) a periodic basis not to exceed two refueling cycles, 2) a process for evaluation and disposition of proposed facility changes for items impacting the PRA model, and 3) if any PRA error is identified that significantly impacts RICT calculations, corrective actions shall be identified and implemented as soon as practicable in accordance with the station corrective action program. RG 1.200, Rev 2, section C.1.4, "PRA Development, Maintenance, and Upgrade," states in part, "The PRA results used to support an application are derived from a PRA model that represents the as-designed, as-built, as-operated plant. Therefore, a process for developing, maintaining, and upgrading a PRA is established." Section C.2, "Consensus PRA Standards and Industry PRA Programs," states in part, "One acceptable approach to demonstrate conformance with regulatory position 1 is to use the national consensus PRA standard," (i.e., ASME/ANS Ra-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"). ASME/ANS RA-Sa-2009, Section 1-5, "PRA Configuration Control," section 1-5.4 states, "the PRA shall be maintained and upgraded such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is being used." In addition, section 1-5.4 states "changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical Requirements Section of each respective part," of the Standard.

Approved SFCPs are included in the Administrative Controls section of the TS requiring changes to frequencies under SFCP to be made in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1. NEI 04-10, Rev 1, Section 4.0, Step 5 requires the PRA technical adequacy to be addressed through RG 1.200 and the ASME PRA Standard. RG 1.200, Rev 2, section C.1.4, "PRA Development, Maintenance, and Upgrade," states in part, "The PRA results used to support an application are derived from a PRA model that represents the as-designed, as-built, as-operated plant. Therefore, a process for developing, maintaining, and upgrading a PRA is established." Section C.2, "Consensus PRA Standards and Industry PRA Programs," states in part, "One acceptable approach to demonstrate conformance with regulatory position 1 is to use the national consensus PRA standard," (i.e., ASME/ANS Ra-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"). ASME/ANS RA-Sa-2009, Section 1-5, "PRA Configuration Control," section 1-5.4 states, "the PRA shall be maintained and upgraded such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is being used." In addition, section 1-5.4 states "changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical Requirements Section of each respective part," of the Standard.