# NRC INSPECTION MANUAL

IRIB

INSPECTION PROCEDURE 71111 ATTACHMENT 13

MAINTENANCE RISK ASSESSMENTS AND EMERGENT WORK CONTROL

Effective Date: July 1, 2026

PROGRAM APPLICABILITY: IMC 2515 Appendix A

CORNERSTONES: Initiating Events
Mitigating Systems
Barrier Integrity

INSPECTION BASES: See IMC 0308, Attachment 2

SAMPLE REQUIREMENTS:

| Sample Requirements | | Minimum Baseline Completion Sample Requirements | | Budgeted Range | |
|---|---|---|---|---|---|
| Sample Type | Section | Frequency | Sample Size | Samples | Hours |
| Risk Assessment and Management[1] | 03.01 | Annual | 10 at one reactor unit sites | 10 to 20 at one reactor unit sites | 65 to 115 hours at one reactor unit sites |
| | | | 12 at two reactor unit sites[2] | 12 to 22 at two reactor unit sites | 75 to 125 hours at two reactor unit sites |
| | | | 14 at three reactor unit sites | 14 to 24 at three reactor unit sites | 85 to 135 hours at three reactor unit sites |
| | | | 7 at Vogtle Units 3 & 4 | 7 to 9 at Vogtle Units 3 & 4 | 35 to 45 hours at Vogtle Units 3 & 4 |

[1] Two samples should include a deeper review of the licensee's PRA configuration control program (one sample for Vogtle 3 & 4).
[2] Also applicable to Vogtle Units 1 & 2.

## 71111.13-01 INSPECTION OBJECTIVES

01.01 To verify that appropriate risk assessments (RAs) and corresponding work controls and risk management actions (RMAs) (including the use of risk-informed completion times (RICTs) if applicable) are implemented during planned and emergent maintenance activities.

01.02 To verify that RAs are based upon an appropriate Probabilistic Risk Assessment (PRA) model, including a configuration control program that ensures the PRA model reflects the actual plant design and licensing bases.

## 71111.13-02 GENERAL GUIDANCE

This inspection procedure (IP) shall be used to examine plant configuration changes associated with scheduled or emergent maintenance activities that may be planned, in progress, or completed. The plant configuration changes to be inspected are those involving structures, systems, or components (SSCs) within the scope of the Maintenance Rule or the limited scope as allowed by Title 10 of the Code of Federal Regulations (10 CFR) 50.65(a)(4)) and certain other risk -significant SSCs.

If a licensee has received NRC approval to adopt risk-informed technical specifications (TS) initiative 4b risk-managed technical specifications (RMTS), the licensee may extend the duration in which the plant may be operating in a single failure vulnerable plant configuration. As described in 10 CFR 50.36(c)(2)(i), limiting conditions for operation (LCO) are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Although 10 CFR 50.36(c)(2) does not describe any specific safety standard for the remedial actions permitted by the TS, both the common standards for licenses in 10 CFR 50.40(a) and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3) state that there must be “reasonable assurance” that the activities at issue will not endanger the health and safety of the public.

Per 10 CFR 50.69(b)(vi), 10 CFR 50.65(a)(4) continues to apply to SSCs that may be excluded from other Maintenance Rule monitoring criteria under 10 CFR 50.69.

Before performing this procedure, develop an understanding of the licensee’s program for conducting RAs and managing risk and become familiar with the associated procedures. For a greater understanding of the regulatory requirements regarding the use and maintenance of RA programs, review Appendices B, C and D of this procedure. Although it is not within the scope of this inspection to perform a programmatic review of the licensee’s 10 CFR 50.65(a)(4) procedures, it would be appropriate to question and bring to the licensee’s attention anything in the procedures discovered during this familiarization that is not clear or appears to be incorrect.

Considering opportunities, risk, and judgment, select appropriate scheduled or emergent work activities for sampling. During plant tours and plant status, look for potential activities that increase plant risk or that may not have been fully evaluated. In addition to potential impacts to reactor safety, consider potential impacts on security and emergency preparedness equipment and the ability to maintain security and emergency preparedness functions.

Licensees that have adopted technical specification RICTs can, under certain circumstances, determine completion times for maintenance and emergent work. Additional information for inspection of RICTs is provided in Appendix A.

For emergent work activities, verify that work schedules and work plans are being followed and that precautions are being taken to preclude affecting adjacent SSCs. Observe equipment lineups and tagging when potential errors could affect other operating systems. When appropriate, verify that redundant components remain operable and available to perform the safety function. Reference IP 71111.04, "Equipment Alignment," for additional guidance. Consider whether potential maintenance errors could initiate an event or affect defense-in-depth when selecting work activities to review. When work activities involve the replacement of safety-significant parts, verify the appropriateness of using commercial-grade parts. Limit the review to emergent work activities that could cause an initiating event, affect the functional capability of mitigating systems or barrier integrity, or affect the ability to accomplish security or emergency preparedness functions.

Each quarter, ensure that a portion of the inspection effort is directed at conducting a routine review of problem identification and resolution activities using IP 71152, "Problem Identification and Resolution (PI&R)."

## 71111.13-03 INSPECTION SAMPLES

03.01 <u>Risk Assessment and Management Sample</u>

**a. Verify that RAs are appropriately performed to address planned or emergent plant configuration changes.**

<u>Specific Guidance:</u>

1. Verify the performance of RAs when required by 10 CFR 50.65(a)(4), with emphasis on higher safety/risk-significant configurations and in accordance with licensee procedures, promptly before emergent work and before changes in plant configuration for maintenance activities, including preventive maintenance, surveillance, and testing, during all modes of plant operation.

2. For emergent work, verify that the licensee performs the RA (to the extent practicable and commensurate with safety) before changing the plant configuration further but, in any case, promptly and to the extent practicable concurrently with, but without delaying, plant stabilization and restoration.

3. Verify by walkdowns that work activities do not introduce new configuration risk, such as by breaching fire, flooding, or security barriers or blocking sprinklers, fire hose stations, or security response equipment, and that they do not introduce temporary systems that create flooding hazards, violate electrical separation, or otherwise present new risk.

b. **Verify that RAs are complete and accurate.**

<u>Specific Guidance:</u>

1. Verify the accuracy and completeness of the information considered in the RA.

2. Verify that the RA tool is appropriately used, i.e., that the licensee uses it in a manner consistent with (1) its capabilities and limitations, (2) plant conditions and evolutions, (3) external events and containment status, and (4) licensee procedures.

For select samples, review the licensee's PRA configuration control program as necessary to ensure the RA reflects actual plant configuration, especially in areas where modifications to SSCs have recently occurred. See Appendix B for more information. Engage the licensee when necessary to ensure that inadequate RAs are promptly addressed.

3. For completed work for which the normal plant configuration has been restored, the licensee may still need to perform (or correctly re-perform) an omitted (or inadequate) RA, or the NRC may need to independently evaluate the configuration in question, if possible, in order to determine the associated change in plant risk for significance determination purposes.

c. **Verify that appropriate work controls or RMAs are implemented in response to RAs.**

Specific Guidance:

1. Verify that the licensee recognizes and enters as applicable the appropriate licensee-established risk category or band according to RA results and licensee procedures.

2. Verify that normal work controls or RMAs are promptly and effectively implemented as required, commensurate with the risk band in effect and in accordance with licensee procedures.

3. Verify that RMAs are effectively implemented in the plant and remain implemented over the course of the entire required period.

4. Verify that the key safety functions for the plant mode of operation are preserved.

5. Re-verify the implementation of RMAs (or different RMAs) that may now be required by licensee procedures following performance (or re-performance) of previously omitted (or inadequate) RAs.

6. During emergent work (combined with scheduled work in progress or alone), verify that the licensee takes actions to minimize the probability of initiating events, maintain the functional capability of mitigating systems, and maintain barrier integrity.

7. Review emergent work-related activities such as troubleshooting, work planning and scheduling, establishing plant conditions and aligning equipment, tagging (clearances), temporary modifications, and equipment restoration to ensure that the plant is not placed in an unacceptable configuration (including violation of technical specifications).

## 71111.13-04 REFERENCES

IMC 2515, Appendix A, "Risk-Informed Baseline Inspection Program"

IMC 0308, Attachment 2, "Technical Basis for Inspection Program"

Cross Reference of Generic Communications to IP 71111.13 and Inspection Resources, available at Microsoft Power BI (powerbigov.us) (nonpublic)

Maintenance Effectiveness:
http://www.nrc.gov/reactors/operating/ops-experience/maintenance-effectiveness.html

Operating Experience: https://usnrc.sharepoint.com/teams/NRR–Operating-Experience-Branch/OpE%20Hub/index.aspx (nonpublic)

END

Appendices:

Appendix A: "Risk-Informed Technical Specifications (TS) Initiative 4b Risk-Managed Technical Specifications (RMTS) Guidelines"
Appendix B: "Probabilistic Risk Assessment Configuration Control Sample Activity"
Appendix C: "PRA Configuration Control Background"
Appendix D: "Operating Experience Background"

Attachment:

Attachment 1: Revision History Table

## Appendix A: Risk-Informed Technical Specifications (TS) Initiative 4b Risk-Managed Technical Specifications (RMTS) Guidelines

### 71111.13A-01 OBJECTIVE

The objective of this Appendix is to support the review of licensees' implementation of the risk -informed TS initiative 4b. Some licensees have been issued license amendments that revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when LCOs are not met. The licensees' submittals were developed using guidance provided in Technical Specifications Task Force (TSTF) Traveler TSTF-505. The U.S. Nuclear Regulatory Commission (NRC) issued a model safety evaluation (SE) approving use of the TSTF-505, however information concerning individual licensees' approvals are documented in the plant-specific safety evaluation report (SER).

### 71111.13A-02 INSPECTION GUIDANCE

a. It is recommended that inspectors who perform some portion of this inspection procedure increase their general familiarity with the licensee's risk model, configuration risk management program, and the plant specific SER issued for the implementation of the RMTS 4b initiative. During the licensee's first implementation of a RICT following approval, the inspector should be aware of and verify any limitations or conditions outlined in the SER.

b. It is also recommended that the inspector become familiar with the plant specific standardized plant analysis risk (SPAR) model to gain insights on equipment configurations and risk important equipment coincident with application of a RICT. Assistance with the use of the SPAR model can be obtained from a Senior Reactor Analyst or Risk Analyst.

c. If the RICT has significant margin (more than one order of magnitude) from the incremental core damage probability thresholds of E-6 for risk management action time and RICT, then this should be taken into consideration when prioritizing inspection and engagement. Stated differently, if the calculation even with significant non-conservatisms added to the calculated value would remain in the normal work controls level, this condition may not warrant inspection effort. On the contrary, if the licensee has or plans to use RICTs using newly developed PRA methods or changes to key assumptions, such RICTs should receive priority in sampling.

d. If a RICT is or has been in effect, verify equipment capable of performing the (specified) safety function of the inoperable equipment remains OPERABLE*. If the licensee credits PRA functionality of inoperable equipment, verify that the functionality and associated RICT are consistent with licensee procedures. Verify that all the constraints specified in the TS Administrative Section and individual technical specifications, as applicable, are met. If required, verify additional RMAs are promptly and effectively implemented in accordance with licensee procedures.

   *Note that if the plant specific TS for the RICT has included loss of function, equipment to perform the SSC must satisfy the additional constraints on loss of function that are specified in the TS Administrative Controls.

e. Verify that the licensee is using the “zero-maintenance” state in their PRA model; (delta-core damage frequency and delta-large early release frequency would be artificially lowered in the calculations if this is not true); this can be easily done by reviewing the cutsets and verifying no test & maintenance terms exist in either the base results or the ‘non-conforming’ results.

f. For emergent conditions, ensure the licensee takes appropriate actions to account for the change in plant configuration. For example, if a high degree of confidence cannot be established that there is no Common Cause failure that could affect the redundant components prior to exceeding the completion time, the RICT shall account for the increased possibility of common cause failure (CCF) by either: 1) numerically accounting for the increased possibility of CCF in the RICT calculation, or 2) implementing additional RMAs not already credited in the RICT calculation.

g. Consider performing a partial equipment walkdown for SSCs that are modeled in the licensee’s PRA and which are supposed to be functional/available during the RICT; the goal is to verify that no risk-significant, credited equipment is in fact unavailable with the licensee unaware of it.

h. Verify that where equipment declared as “inoperable” is being classified as “functional” for purposes of a RICT calculation, the reasoning behind such a consideration is justified in the documentation of the RICT assessment. This reasoning should be credible and technically defensible.

i. Verify that a total loss of function has not inadvertently been created. See Nuclear Energy Institute (NEI) 06-09 Rev. 0-A, Section 3.2.2.

j. Verify that the licensee tracks the risk associated with all entries beyond the front-stop completion time to ensure cumulative risk remains within Regions II or III of Figures 3 and 4 of Regulatory Guide (RG) 1.174 for a 52-week period. See NEI 06-09 Rev. 0-A, Section 3.2.5.

## 71111.13A-03 REFERENCES


Final Revised Safety Evaluation by the Office of Nuclear Reactor Regulation Technical Specifications Task Force Traveler TSTF-505, Revision 2 (ML18267A259)

Licensee Safety Evaluation Report (SER) for the license amendment adopting RMTS 4b

NEI (formerly Nuclear Management and Resources Council (NUMARC)), NUMARC 93-01, Revision 4F, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” (ML18120A069)

NEI 06-09 Revision 0-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines” (ML12286A322)

NRC Regulatory Guide (RG) 1.174, Rev. 3, “An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis” (ML17317A256)

Technical Specification Task Force (TSTF) comments on Draft Safety Evaluation for Traveler TSTF-505, "Provide Risk-Informed Extended Completion Times" and Submittal of TSTF-505, Revision 2 (ML18183A493)

Cross Reference of Generic Communications to IP 71111.13 and Inspection Resources, available at Microsoft Power BI (powerbigov.us) (non-public)

END

## Appendix B: Probabilistic Risk Assessment Configuration Control Sample Activity

### 71111.13B-01 INSPECTION OBJECTIVE

01.01 Provide additional inspection guidance for the review of licensee Probabilistic Risk Assessment (PRA) Configuration Control (PCC) programs.

### 71111.13B-02 BACKGROUND

02.01 Reactor Oversight Process PCC Framework

The NRC established a working group (WG) to address an identified gap in the reactor oversight process (ROP) regarding PCC within risk-informed programs (RIP), using a balanced and graded approach. Specifically, upon NRC approval of a license amendment and completion of implementation inspection if performed, there is adequate assurance that the PRA model accurately reflects the as-built, as-operated plant. However, there is no subsequent follow-up inspection activity to provide continued assurance of PCC and plant changes to ensure that the PRA continues to accurately reflect the as-built, as-operated plant.

Refer to Appendix C, "PRA Configuration Control Background," for additional information related to the regulatory requirements for PCC.

As part of an effort to develop PCC inspection guidance, the WG conducted eight voluntary tabletop visits to different facilities with approved RIP. These tabletops were conducted in partnership with the licensees to understand the licensees' implementation of PCC programs and to assist in optimizing future inspection guidance in this area.

02.02 Operating Experience

The results of the tabletop visits provided valuable insights into the implementation of PCC programs and facilitated the WG's inspection guidance, which at present includes this inspection guidance. For additional specific details on the tabletops and NRC staff observations, refer to the memorandum "Summary of the U.S. Nuclear Regulatory Commission Staff Observations from the Probabilistic Risk Assessment Configuration Control Tabletop Site Visits" (ADAMS Accession No. ML23136A565). High-level observations from the memo are included in Appendix D, "Operating Experience Background."

### 71111.13B-03 INSPECTION GUIDANCE

03.01 It is recommended that inspectors perform some portion of this appendix during the conduct of IP 71111.13 inspection activities, to increase their familiarity with the licensee's PCC model, and obtain a deeper understanding of the site risk management program. Inspectors should also consider using this inspection guidance to support other baseline inspection activities if risk management insights are desired.

a. Prior to sample selection, inspectors should, in coordination with regional Senior Reactor Analysts (SRA), consider the following:

1. Industry and site-specific licensee operating experience concerning implementation of advanced RIP, such as Risk-Informed Completion Time (RICT), National Fire Protection Association (NFPA) 805, 10 CFR 50.69, and Surveillance Frequency Control Program (SFCP).

2. Approval of RIP license amendments and issuance of safety evaluations. If the safety evaluation or model of record (MOR) update has been made within the last 3 years, then the MOR is most likely to have not had significant updates since the initial approval. Inspectors should prioritize sample selection for significant updates to RIP.

3. Timing of last MOR update. Inspectors should prioritize sample selection for MOR updates that have not been previously reviewed.

b. In general, when using this guidance, inspectors should be aware of the processes the licensee has in place to maintain PCC, specifically the following:

1. Does the licensee have built in processes and measures to ensure that plant changes are communicated to the personnel responsible for PRA maintenance (i.e., are there signoffs and reviews during modification reviews or are the changes managed informally between engineering and the personnel responsible for PRA maintenance)?

2. What process does the licensee use to manage and track PRA changes (i.e., Corrective Action Program (CAP), document control, etc.)?

c. Communicate any issues of concern that warrant screening, even for minor issues, to the regional SRA. All issues will be dispositioned, or monitored, in conjunction with the Significance Determination Process.

03.02 Verify processes and procedures are in place to ensure the PRA program is being maintained to support risk-informed decisions. Inspectors should consult with regional SRAs or other subject matter experts to determine what samples are appropriate. When using this inspection guidance, it is understood that selected items or changes for review likely have not had sufficient time to have been fully incorporated into the MOR by the PRA update process. However, they may have been incorporated into a working PRA model. The intent is to verify that the licensee has a robust process in place to reasonably ensure PCC activities.

a. As applicable, with respect to the selected sample, verify the following processes and procedures are in place to monitor and update the PRA.

1. Review PCC-related administrative procedures to ensure that PCC processes are in place and are being followed.

2. Review a sampling of the most recent MOR, the MOR inputs, and most recent PRA maintenance data/notebooks, or similar, to ensure the licensee is performing the expected maintenance as required.

3. Review the licensees’ PRA tracking and evaluation processes to ensure that any significant changes to the plant (i.e., modifications, procedure changes, etc.) are being communicated to the PRA group for evaluation for inclusion in the PRA and MOR.

4. Verify the licensee is evaluating cumulative changes to the PRA and to the plant. This can be accomplished by reviewing the licensee’s change log or equivalent.

5. Verify the licensee maintains adequate tracking and accounting of PRA maintenance and PRA upgrades, and that MOR changes are being properly evaluated and tracked as either maintenance or upgrades. If any PRA upgrades have been performed, verify the licensee has completed a peer review for that specific PRA upgrade, in accordance with the PRA standard and industry processes.

6. Verify that the licensee is working to address and update any PRA evaluation backlogs such that there are no long-standing PRA related issues that need to be evaluated and addressed, which may include outstanding peer review findings. If there are long-standing unaddressed issues, determine why those have not been addressed and ensure there is no significant impact.

b. As applicable, with respect to the selected sample and IP being performed, verify processes and procedures are in place to monitor PRA inputs and to collect new information and data (e.g., updated industry failure rates).

1. Verify MOR updates are being performed in a timely manner depending on license conditions such as 10 CFR 50.69, Risk-Informed Completion Times (RICT), other commitments, or self-imposed standards. Updates typically should be made within a frequency of every two refueling outages in accordance with licensee procedures.

2. Verify the licensee is performing data gathering and data updates in a timely manner. Review PRA data notebooks, or similar, to ensure that data is relatively current, and if not, determine if there is appropriate justification for not performing the data update.

c. If applicable, review modifications to the plant that appear to be risk significant. For reference, “risk significant” may include risk significant SSCs, or items such as success path changes, added event tree tops, fault-tree updates, and other means as necessary. Other options for consideration may include recent licensee changes to human error probabilities, changes to minimum equipment and or success criteria, changes to importance measures (i.e., Birnbaum, Fussell-Vesely, Risk Achievement Worth, etc.). Inspectors may utilize Regional SRAs or DRA/APOB Branch for support in determining appropriate significant samples, as necessary. Inspectors may also utilize the Plant Risk Information eBooks (PRIB) for reference.

1. Verify that risk significant modifications and/or procedure changes are included or being tracked in the PRA change log, or equivalent tracking mechanism, to ensure the changes are being reviewed for appropriate evaluation and future inclusion into the in the PRA and the next MOR update.

03.03 Verify processes and procedures were completed appropriately to ensure the PRA program was being sufficiently maintained to support past and current risk-informed decisions. Inspectors should utilize regional or headquarters SRAs for support in determining which samples are appropriate for application of this activity, as necessary. When performing sections of this procedure, it is preferable to select items or changes for review that have had sufficient time to have been processed into the MOR by the PRA update process. The intent is to verify that the licensee has a robust process in place to reasonably ensure PCC activities.

a. As applicable, with respect to the selected sample being performed, verify processes and procedures were followed to monitor and update the PRA.

   1. Review significant PRA updates or changes to PRA inputs, specifically risk-significant systems such as those implementing RIP (e.g., RICT).

   2. For insights on possible samples for this procedure, review one of the following: PRIB, Mitigating System Performance Index input recalculations, licensee PRA MOR change logs, PRA system notebooks, or equivalent tracking systems. Preferably any changes reviewed will already be processed and in the current MOR.

   3. Review a sampling of past MOR inputs and PRA Maintenance data/notebooks to ensure the licensee has been performing the expected maintenance to data inputs as required and that the changes have been evaluated and processed into the current MOR (e.g., updated industry failure rates). If the sample involves repetitive failures of equipment, consult the regional SRA to assist in the review and evaluation of licensee PRA procedures.

b. As applicable, with respect to the selected sample being performed, verify processes and procedures were followed to ensure that the PRA was maintained with the as-built, as-operated plant.

   1. Review a sampling of past significant modifications (i.e., success path changes, added event tree tops, or new fault-trees, etc.) to the plant to ensure that they were properly screened for inclusion in the PRA and have been included in the appropriate MOR update. This can be accomplished by reviewing a sampling of past significant modifications and comparing those to the PRA change log to ensure the items were appropriately reviewed and evaluated.

   2. Review the licensee's processes to ensure that any significant changes to the plant's procedures, specifically procedure changes that contain human reliability analysis (HRA) changes with time constraints, etc., were communicated to the PRA group for evaluation for inclusion in the PRA and MOR.

   3. Verify that any significant changes to the operating procedures (i.e., Emergency Operating Procedures (EOPs), HRAs, etc.) have been processed through the PRA update process with the appropriate timeliness (i.e., every two refueling outages frequency). If not, determine if there is appropriate justification for not performing the update.

## 71111.13B-04 REFERENCES

These references may include pre-decisional information contained on NRC internal websites. Once the agency has formally evaluated an OpE issue and has determined that it meets the criteria for agency action, the NRC communicates the issue to the public and the industry through one or more appropriate methods (e.g., generic communication, rulemaking public comment periods, etc.).

04.01 IP 37060, "10 CFR 50.69 Risk--Informed Categorization and Treatment of Structures, Systems, and Components Inspection"

04.02 “Summary of the U.S. Nuclear Regulatory Commission Staff Observations from the Probabilistic Risk Assessment Configuration Control Tabletop Site Visits” (ML23136A565)

## Appendix C: PRA Configuration Control Background

### A.1 PRA 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 encouraged the use of PRA in all regulatory matters and 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."

PRA Quality was first described in SECY-00-0162, "Addressing PRA Quality in Risk-Informed Activities" (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. SECY-04-0118, "Plan for Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality" (ML041470505), defined PRA Quality as in Regulatory Guide (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, (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.

Regulatory Guide 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, and
- the technical attributes and characteristics for a full-scope PRA.

Regulatory Position 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 were 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 were included. The RG endorsed, with exceptions (i.e., clarifications and qualifications), the American Society of Mechanical Engineers (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. Nuclear Energy Institute (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.

NOTE: ASME Standards may be obtained through the NRR ROP Digital City SharePoint site (non-public), under the “Guidance” section link for “Codes and Standards.” ASME codes can be found within the “Accuris (Formerly IHS) Codes and Standards” link (non-public). A login is required for access.

Regulatory Guide 1.200, Revision 1, (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 addenda 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.

Regulatory Guide 1.200, Revision 2 (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

Regulatory Position 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.

In 2007, the NRC issued Regulatory Issue Summary (RIS) 2007-06, “Regulatory Guide 1.200 Implementation” (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.

Regulatory Guide 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 3 (ML20238B871), introduced the term PRA Acceptability which had been synonymous with previously used terms such as PRA Quality and PRA technical adequacy. Regulatory Guide 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
- Plant representation and PRA configuration control

Regulatory Position C.2.2 added guidance for peer review of upgrades or any newly developed methods. 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, (ML19231A182), in its entirety as a means of satisfying the peer-review requirements for the ASME/ANS RA-Sa-2009 PRA standard.

A.2 Regulatory Requirements for PCC

Risk-informed programs (RIP) require PRA Configuration Controls (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 RIP. 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 (SSC) 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 Risk-Informed Completion Time (RICT) program

- TSTF 425, “Relocate Surveillance Frequencies to Licensee Control,” Revision 3, or the Surveillance Frequency Control Program (SFCP)

Specific regulatory requirements for each RIP are detailed below:

a. 10 CFR 50.69, Risk-Informed Categorization and Treatment of SSC for Nuclear Power Reactors

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.

b. 10 CFR 50.48, Fire Protection, NFPA 805

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, regulatory position C.1 and C.2 are an acceptable way to demonstrate PRA Technical Adequacy per Revision 2 or PRA Acceptability per Revision 3.

c. TSTF–505, Risk-Informed Completion Time (RICT)

Approved RICT programs are included in the Administrative Controls section of the Technical Specifications (TS) which required the program to be implemented IAW NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." NEI 06-09-A, Revision 0, Section 2.3.4, "PRA Technical Adequacy," item 2, required the PRA to be reviewed to the guidance of RG 1.200, Revision 0, for a PRA which meets Capability Category (CC) 2 for the supporting requirements 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.

Regulatory Guide 1.200, Revision 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.

d. TSTF 425, "Relocate Surveillance Frequencies to Licensee Control," Revision 3, or the Surveillance Frequency Control Program (SFCP)

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, Revision 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, Revision 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.

## Appendix D: Operating Experience Background

This appendix includes observations from voluntary tabletop visits. For additional specific details on the tabletops and NRC staff observations, refer to the memorandum “Summary of the U.S. Nuclear Regulatory Commission Staff Observations from the Probabilistic Risk Assessment Configuration Control Tabletop Site Visits” (ML23136A565).

### B.1 Generic Tabletop Observations

The following are generic high-level observations available in the memo identified above. Generally, the NRC found that the licensees are implementing their PCC programs in a sufficient manner to support RIP. However, the NRC noted vulnerabilities in the areas of documentation and an over-reliance on knowledge-based programs. Specifically, the NRC observed that for most facilities, the engineering input monitoring process is well defined. However, the process for monitoring operations, maintenance, component performance monitoring, and industrywide operational experience appeared to be informal. 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 licensee PRA staff to evaluate. This led to some examples where failure events were screened from parameter updates without sufficient justification, potential failure modes were not modeled, and some industrywide operating experience lacked licensee evaluation. The NRC also noted some instances of parameter data not being updated in a timely manner.

### B.2 Specific Tabletop Observations

For additional information related to the observations below, review the tabletop memo above. Since the tabletops were voluntary, these examples were potential issues only, not documented findings or violations. Observations that were identified by the NRC were provided to the licensee for further evaluation and disposition. During this inspection activity, such observations would be evaluated for importance in accordance with the significance determination process.

a. NRC reviewed a maintenance log item, which documented the vulnerability to a major flood from the turbine building to the auxiliary feedwater (AFW) rooms. Specifically, a plant modification to the turbine driven AFW (TDAFW) pump room created a common drain between the TDAFW and the motor-driven AFW (MDAFW) pump rooms. The licensee failed to consider the vulnerability of the float check valves in the MDAFW pump room; and the potential failure of these valves to close was not included in the flooding hazard group model because it was not considered significant. The licensee entered this issue into its CAP for further evaluation. This is an example of components that protect from a major flood that were potentially not appropriately representing the as-built, as-operated plant following a plant modification.

b. NRC reviewed an engineering change (EC) for installation of two non-safety-related diesel generators (DGs) capable of supplying a safety-related 4 kV bus to either unit, for a dual unit site. The DGs were synchronized onto a 4 kV bus through a programmable logic controller (PLC) that required both engines to be operating and their individual output breaker closed before their tiebreaker was closed to the bus. The system notebook for the DGs established one DG supplying 4 kV power to one 4 kV safeguard bus as a success criterion. For one DG to operate, EOPs direct operators to override the PLC and take additional steps to restore power to a safety train in the station blackout

(SBO) unit. NRC reviewed the DGs representation in the PRA model and found that the human failure event was not included for the PLC override. The licensee entered this issue into the CAP for further evaluation. This is an example of data screened out for PRA parameter estimation without a documented justification.

c. NRC 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 AFW flow to 2 of 4 Steam Generators for the first 4 hours of the event. The alternate water source from essential service water was connected to the suction of the TDAFW pump, upstream of the suction check valve. NRC staff noted that the AFW system notebook failed to assume or model a flow diversion from the supply lines between the MDAFW and the TDAFW pumps, given a failure of one of either of the AFW pumps. NRC staff confirmed by reviewing the AFW fault tree that the failure to close the suction check valve was not modeled. The AFW system notebook did not include a documented basis to address screening out this vulnerability. The licensee entered this issue into the CAP for further evaluation. 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 without adequate documented justification to screen out a failure event for the internal events model.

d. NRC reviewed a PCC evaluation of the open phase condition (OPC) design vulnerability in the electric power systems as an input to industrywide operational history. OPC as communicated to industry in NRC Bulletin 2012-01, "Design Vulnerability in Electric Power Systems" (ML12074A115), was ultimately resolved by industry with NEI 19-02, Guidance for Assessing OPC Implementation Using Risk Insights" (ML19172A086). NEI 19-02 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 OPC. Following the guidance of NEI 19-02 does not relieve licensees of their PCC requirement to evaluate OPC as an industrywide operational history input for potential maintenance or update of the PRA. The licensee had not performed an OPC PCC evaluation and entered this issue into the CAP for further evaluation. This is an example of the model potentially not representing the as-built, as-operated plant.

e. NRC reviewed the latest data update performed by the licensee. NRC found that the latest 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 Risk-Informed Completion Time (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. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69 (e)(1) requires PRA updates to be reviewed in a timely manner but no longer than once every two refueling outages. The PCC evaluation concluded that the change in core damage frequency to the model of record (MOR) was small. However, an evaluation of the potential impact to the RICT calculation was not conducted. The licensee entered this issue into the CAP for further evaluation. This is an example of the MOR not potentially representing the as-built, as-operated plant.

Attachment 1: Revision History for IP 71111.13

| Commitment Tracking Number | Accession Number Issue Date Change Notice | Description of Change | Description of Training Required and Completion Date | Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information) |
|---|---|---|---|---|
| N/A | 3/13/2007 | Revision history reviewed for the last four years - no generic requirements incorporated during this period. | None | N/A |
| [C1] SRM M050426 | ML070240479<br>03/23/07<br>CN 07-011 | This IP is revised to incorporate inspections for the offsite power system and the alternate AC power source. | Yes<br>12/13/2006 | ML070680061 |
| N/A | ML072910050<br>01/31/08<br>CN 08-005 | This IP is revised to incorporate results of the ROP realignment in 2007, the references were updated to reflect the deletion of TI 2515/165 in CN 07-28 and minor typographical errors were corrected. | None | N/A |
| N/A | ML092380216<br>11/16/09<br>CN 09-027 | This IP is revised to incorporate results from the ROP realignment in 2009. Recommendations from ROPFF 71111.13-1360 and 71111.12-1407 were also added. The table in the General Guidance section was deleted and editorial corrections were made. | None | ML093010336 |
| N/A | ML11201A172<br>10/28/11<br>CN 11-025 | This revision modifies the resource estimate to reflect the 2011 ROP Realignment. | None | |
| N/A | ML15023A099<br>02/03/16<br>CN 16-005 | Revision 3 to RG 1.160 and Revision 4A to NUMARC 93-01 were recently issued. IP 71111.12 has been revised to update references to the new revision numbers.<br><br>Changes made in accordance with ROP Enhancement Project (see ML14017A381). | | ML16007A383<br><br>71111.13-1951<br>ML16033A375<br>71111.13-2135<br>ML16033A384 |

| Commitment Tracking Number | Accession Number Issue Date Change Notice | Description of Change | Description of Training Required and Completion Date | Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information) |
|---|---|---|---|---|
| N/A | ML17194A934<br>12/20/17<br>CN 17-030 | Adding inspection of RICT times and risk management actions. These are a subset of maintenance activities and are included in the samples. This change includes information to facilitate inspection of RICT and PRA functionality.<br><br>Streamline IP formatting | | ML17205A097<br>71111.13-2261<br>ML17205A261 |
| | ML19198A075<br>DRAFT<br>CN 19-XXX | Made publicly available to discuss at the July 31, 2019, public meeting. | | |
| N/A | ML19197A096<br>11/13/19<br>CN 19-035 | Additional guidance for inspecting RICTs added for clarity. | | ML19210C938<br><br>71111.13-2358<br>ML19301A004 |
| N/A | ML20238B972<br>10/05/20<br>CN 20-046 | Revisions are made to add inspection samples specifically for Vogtle 3 & 4 as identified in SECY-20-0050, “Planned Revisions To The Baseline Inspection Program For The AP1000 Reactor Design,” (ML20058F491). | None | ML20239A737 |
| | ML22154A388<br>08/01/22<br>CN 22-015 | Implemented recommended changes as a result of ROP Enhancement efforts. | | ML22175A147 |

| Commitment Tracking Number | Accession Number<br>Issue Date<br>Change Notice | Description of Change | Description of Training Required and Completion Date | Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information) |
|---|---|---|---|---|
| | ML25357A217<br>05/01/26<br>CN 26-017 | Adjusted inspection samples/hours and added an inspection activity to conduct a review of select portions of site PRAs to verify the information contained in the PRAs reflect actual site configuration. These revisions were recommended as a result of the ADVANCE Act Section 507 Report to Congress that discussed the revision of the ROP Baseline Inspection Program and are summarized in ML25247A050. | N/A | ML25274A088 |