NRC INSPECTION MANUAL

IRIB

INSPECTION PROCEDURE 37060

10 CFR 50.69 RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS INSPECTION

Effective Date: 10/18/2022

PROGRAM APPLICABILITY: IMC 2515 App C

37060-01 INSPECTION OBJECTIVES

01.01 To verify that the licensee's programs and procedures have properly incorporated the license amendment under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," as approved by the U.S. Nuclear Regulatory Commission (NRC) staff, described in the plant safety evaluation and Updated Final Safety Analysis Report (UFSAR), and documented in the staff's Safety Evaluation (SE).

01.02 To verify that the licensee properly implements the 10 CFR 50.69 categorization process consistent with the 10 CFR 50.69 regulatory requirements for the structures, systems, and components (SSCs) subjected to the process.

01.03 To verify that the licensee properly implements alternate treatment requirements for those SSCs evaluated using the 10 CFR 50.69 categorization process.

37060-02 INSPECTION REQUIREMENTS

02.01 Review of the Licensee's Programs and Procedures.

The inspector shall review the licensee's programs and procedures to ensure that the procedures fully describe the categorization and treatment process for SSCs as described in its UFSAR and as required by 10 CFR 50.69. Specifically, inspectors shall verify the following aspects of the licensee's programs and procedures:

a. Verify the process for categorization of SSCs into risk-informed safety classes as described in licensee procedures is consistent with the categorization process approved by the NRC.

b. Verify license conditions, including updates to the probabilistic risk assessment (PRA) model from the NRC SE are properly incorporated into licensee procedures and programs.

c. Verify the licensee maintains the plant specific PRA models, including severe accident scenarios, to reflect the as-built and as-operated plant.

d. Verify the procedure containing the description of the integrated, systematic process to determine the functional importance of SSCs is consistent with the description in the license amendment application and the NRC's SE.

e. Verify licensee procedures describe how the licensee maintains defense-in-depth.

f. Verify licensee procedures describe the basis for the acceptability of the evaluations to be conducted to provide reasonable confidence that the licensee is maintaining sufficient safety margins.

g. Verify licensee procedures require evaluation of entire systems and structures, not just selected components within a system or structure.

h. Verify licensee procedures require SSCs categorization to be reviewed by an integrated decision-making panel (IDP) that is staffed with expert, plant-knowledgeable members whose joint expertise includes, at a minimum: PRA, safety analysis, plant operation, design engineering, and system engineering.

i. Verify licensee procedures describe the monitoring and evaluation of the treatment applied to RISC-1 and RISC-2 SSCs.

j. Verify licensee procedures describe alternate treatment process to ensure, with reasonable confidence, that RISC-3 SSCs will continue to be capable of performing their safety-related functions under design-basis conditions.

k. Verify licensee procedures require feedback and process adjustments.

02.02 Review of the Licensee's 10 CFR 50.69 Program Implementation

The inspector shall select three to five systems that the licensee evaluated using its approved 10 CFR 50.69 categorization process. The inspector shall verify the licensee categorized the selected systems' SSCs in accordance with procedures and regulatory requirements as follows:

a. Verify the licensee has implemented their integrated, systematic categorization process in accordance with procedures and regulatory requirements.

b. Verify the licensee preserved defense-in-depth and safety margin for SSCs categorized as RISC-3.

c. Verify the licensee performed RISC-3 SSC categorization evaluations in accordance with procedures.

d. Verify the license performed evaluations of the entire systems and structures

e. Verify the licensee's IDP was staffed in accordance with procedures and regulatory requirements.

f. Verify the licensee's treatment and evaluation of RISC-1 and RISC-2 SSCs supports the functions credited in the PRA model

g. Verify the licensee’s alternate treatment applied to RISC-3 SSCs provides reasonable confidence that these components will continue to be capable of performing their safety related functions under design basis conditions.

h. For the selected SSCs, verify that the licensee is implementing the requirements for program documentation, change control, and records in 10 CFR 50.69(f).

i. For the selected SSCs, verify that the licensee is implementing the reporting requirements in 10 CFR 50.69(g).

02.03 Problem Identification and Resolution

For the systems selected, the inspector shall review the licensee’s problem identification and resolution system to verify that conditions that would prevent RISC-3 SSCs from performing their safety-related function have been corrected. Additionally, the inspector shall verify the licensee appropriately performs common cause reviews of RISC-3 SSCs deficiencies such that risks are identified and assessed prior to events that could invalidate the licensee’s SSC categorization process.

Additionally, review the licensee’s past audits and self-assessments performed on the implementation of the 10 CFR 50.69 program to ensure that it took adequate corrective actions from these audits.

02.04 Review of Licensee’s Feedback and Process Adjustments

Following two refueling outages after system categorization, the inspectors shall select three to five systems to verify that the licensee is implementing the requirements for feedback and process adjustment in 10 CFR 50.69(e). Specifically, inspectors shall verify the following aspects of the licensee’s feedback process:

a. Verify the licensee reviews changes to the plant, operational practices, operating experience, and updates the PRA and categorization or treatment process in a timely manner.

b. Verify the licensee is monitoring the performance of RISC-1 and RISC-2 SSCs and has made adjustments, as necessary, to the categorization or treatment so that the categorization process remains valid.

c. Verify the licensee has considered data collected for RISC-3 SSCs and has made adjustments, as necessary, to the categorization or treatment so that the categorization process remains valid.

37060-03 INSPECTION GUIDANCE

The team should become familiar with the following documents before starting their inspection of the licensee’s implementation of the 10 CFR 50.69 license amendment:

a. Licensee’s 10 CFR 50.69 license amendments and the NRC’s safety evaluation of the license amendment application.

b. *Federal Register* Notice (69 FR 68008, dated November 22, 2004) – Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Plants (ADAMS Accession No. ML042960073).

c. RG 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to Their Safety Significance,” Revision 1 (ML061090627).

d. NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” Revision 0 (ML052910035).

e. Licensee’s procedures used to establish and to provide guidance on how to implement the risk-informed categorization and treatment of SSCs.

f. 10 CFR 50.69 Inspection Guidance and Resources (ML21273A302, non-public)

In reviewing these documents, the inspector should become familiar with the plans and programs specified by the licensee in the license amendment application and relied on by the NRC staff in granting the license amendment in the safety evaluation report. For example, the inspector should understand the categorization process and the treatment programs for RISC-1, RISC-2, RISC-3, and RISC-4 SSCs. The inspector should discuss the planned inspection with the cognizant NRC Headquarters staff to identify any specific areas of inspection that might be warranted.

03.01 Review of the Licensee’s Programs and Procedures.

a. Before a licensee can implement the requirements in 10 CFR 50.69, the NRC must approve the categorization process through licensee’s submittal of a license amendment under 10 CFR 50.90. The NRC’s approval of a license amendment is based on a determination that the categorization process satisfies the requirements of 10 CFR 50.69(c). The licensee must describe the SSC categorization process for risk-informed safety class (RISC)-1, RISC-2, RISC3, and RISC-4 in site procedures that are consistent with the categorization process approved in the NRC SE.

   The review and approval of the licensee categorization process is a one-time process approval. The approved categorization process can be applied to any SSC in the plant. In addition, the licensee is not required to return to the NRC for review of the categorization process output provided that its process remains within the scope of the NRC’s safety evaluation.

   The licensee should have implementing procedure(s) for properly categorizing each component using 10 CFR 50.69. The plant procedures should be consistent with the NRC-approved categorization process as described in the licensee’s UFSAR and sufficiently detailed to provide assurance that the licensee will properly categorize components. Additionally, licensee procedures should reflect approved methodologies as specified in the license’s amendment request and NRC SE.

   The description of the categorization of SSCs into RISC-1, RISC-2, RISC-3, and RISC-4 categories should include the process to categorize the safety-significance of components based on the active (mechanical and electrical) functions of a component, the passive functions of a component (pressure boundary), and, for those components that are modeled in the PRA, the importance of the component to the risk estimates.

Different portions of an SSC may be assigned different categorization levels. For example, the motor operator of a normally closed motor operated valve may be assigned different safety significance than the valve body if the safety-significant function is to remain closed. The licensee's process for categorizing portions of SSCs that can have different categorization levels should ensure that the process is consistent with the license amendment application and the staff's findings in the NRC safety evaluation.

b. The NRC issued 50.69 amendment may contain license conditions requiring the licensee to perform specific updates to procedures and programs. Updates to PRA modeling methods can impact the results of the categorization process. Required updates to licensee procedures and programs should be completed prior to implementation of the categorization of any SSCs.

c. In accordance with 10 CFR 50.69(c)(1)(i), the licensee's PRA is required to be of sufficient quality and level of detail to support the categorization process. The licensee's PRA must model severe accident scenarios resulting from internal initiating events occurring at full power. The NRC's review of the 10 CFR 50.69 submittal will determine whether the requirements in the regulation are satisfied and will determine if the scope, level of detail, and technical adequacy of the PRA is sufficient to support the categorization process. The PRA should be maintained and upgraded, when appropriate, as described in the ASME/ANS PRA Standard endorsed by the latest revision of Regulatory Guide 1.200. All aspects of the integrated, systemic process used to characterize SSC importance must reasonably reflect the current plant design, operating practices and applicable plant and industry operational experience. Inspectors should verify that the PRA maintenance and upgrade procedures requirements are being accomplished by the licensee.

d. In accordance with 10 CFR 50.69(c)(1)(ii) the licensee's categorization process must determine the functional importance of SSCs, using an integrated, systematic process for addressing initiating events (internal and external), and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified include design basis functions and those functions credited for mitigation and prevention of severe accidents. Licensee procedures should describe the integrated, systematic process that is used to perform the categorization of SSCs.

e. In accordance with 10 CFR 50.69(c)(1)(iii) the licensee's categorization process must maintain defense-in-depth. For safety-related components that are determined to be of low risk significance, licensee procedures should provide direction for verifying defense in depth is maintained.

f. In accordance with 10 CFR 50.69(c)(1)(iv) the licensee's categorization process must include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment are small. Licensee procedures should describe the risk sensitivity study that is performed as part of the categorization and evaluation process. The risk sensitivity study should confirm that increases in CDF and LERF are sufficiently small.

g. In accordance with 10 CFR 50.69(c)(1)(v), the licensee's categorization process must categorize an entire system or structure, not just selected components within a system or structure. This required scope ensures that licensees will properly identify and evaluate all system functions (both safety and non-safety related system functions) associated with a system or structure when determining the safety significance of individual components within a system or structure and that they will consider and address the entire set of components (to a reasonable level of detail, e.g., all SSCs depicted on a piping and instrument or a single line diagram) that comprise a system or structure.

h. In accordance with 10 CFR 50.69(c)(2), the licensee is required to review SSC initial categorization by an IDP that will finalize the categorization through the incorporation of evaluation results with risk insights and other traditional information. Because the IDP makes the final determination about the safety significance of an SSC, the requirements in 10 CFR 50.69(c)(2) are necessary to ensure that the panel comprises experienced personnel who possess diverse knowledge and insights in plant design and operation and who are capable of blending deterministic knowledge and risk insights to categorize SSCs.

i. In accordance with 10 CFR 50.69(d)(1) and 10 CFR 50.69(e)(2), the licensee is required to monitor and evaluate the treatment of RISC-1 and RISC-2 SSCs to ensure that these SSCs will perform their safety significant functions consistent with the categorization process assumptions and that the results remain valid. Licensee procedures should describe the method for monitoring and evaluation including the process for making adjustments, as necessary.

j. In accordance with 10 CFR 50.69(d)(2), the licensee shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The licensee must establish treatment that provides reasonable confidence that the RISC-3 SSCs perform their safety-related functions under design basis conditions and is consistent with the assumptions in the categorization process.

   Licensee procedures should describe the inspection and testing, and corrective action that shall be provided for RISC-3 SSCs. Licensee procedures should address that appropriate periodic inspection and testing activities must be conducted for RISC-3 SSCs. Additionally, licensee procedures should provide for corrective actions for conditions that would prevent a RISC-3 SSC from performing its safety-related functions

k. In accordance with 10 CFR 50.69(e), the licensee shall perform a periodic review of the SSC categorization and treatment process. Licensee procedures should establish a process for performing a review of changes to the plant, operational practices, and applicable plant and industry operating experience so that appropriate updates to the PRA and SSC categorization and treatment processes can be identified and implemented, if necessary. Additionally, the licensees review should consider data collected relative to the inspection and testing of RISC-3 SSCs to determine if there have been any adverse changes in performance in excess of the categorization process assumptions.

   The validity of the categorization process relies on the licensee's ability to ensure that it continues to maintain the performance and condition of SSCs credited in the categorization basis. Changes in the level of treatment applied to an SSC might result in changes in the performance or condition of the SSCs. Separately, modifications to system design, changes to operational practices, and plant and industry operational experience may impact categorization process results.

03.02 Review of the Licensee's 10 CFR 50.69 Program Implementation.

a. Many SSCs in the plant will not be modeled explicitly in the PRA. Therefore, the categorization process must determine the safety significance of these SSCs by other means. Because importance measures are not available for use as screening, other criteria or considerations must be used by the IDP to determine the significance. Guidance on how these deliberations should be conducted is included in the NRC regulatory guidance 1.201 and in the industry guidance document, NEI 00-04.

   For the selected SSCs, the inspector should confirm that the licensee properly categorized the active and passive functions and included consideration of initiating events not included in the PRA. If SSCs are separated into subparts or portions, each portion should be properly categorized.

b. The IDP must demonstrate that defense-in-depth is maintained when categorizing SSCs as low safety significant. Defense-in-depth is adequate if the overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure the risk acceptance guidelines are met, and that (1) reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release; (2) system redundancy, independence, and diversity is preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters; (3) there is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design; and (4) potential for CCFs is taken into account.

   For example, the safety-related function of the containment and its systems must not be significantly degraded when SSCs are moved to the RISC-3 category. The Commission's position is that the containment and its systems are important in the preservation of defense-in-depth. Therefore, a licensee should demonstrate that the function of the containment as a barrier is not significantly degraded when SSCs that support the functions are moved to RISC-3 (e.g., containment isolation or containment heat removal systems).

c. For the selected SSCs, the inspector should confirm that the licensee's evaluation performed to satisfy 10 CFR 50.69(c)(1)(iv) demonstrates that moderate variations in the failure probabilities of PRA-modeled components categorized as RISC-3 will result in a small change to core damage and large early release frequency. A quantitative demonstration based on sensitivity studies is required each time the program is expanded to a new system. The methodology used to define moderate variations in the failure probabilities will be defined during the LAR review. The inspector should verify that these calculations are being performed before incorporating new SSCs into the program and that the cumulative results are being compared to the appropriate quantitative guidelines.

   The reliability of RISC-3 SSCs could change due to the reduction in treatment applied to these SSCs as a result of 10 CFR 50.69 implementation. However, it is difficult to establish a cause and effect related to specific changes in treatment and resultant changes in SSC reliability. As a result, the regulation was structured to contain: (1) robust categorization and PRA requirements; (2) requirements to show that the implementation risk is small even if the failure rate of SSCs subjected to reduced treatment increases moderately; (3) a provision to make it clear that the treatment applied to RISC-3 SSCs must maintain the validity of the categorization basis; (4) feedback requirements of 10 CFR 50.69(e) to maintain the validity of the categorization basis; and (5) the high-level RISC-3 requirements designed to maintain reasonable confidence of RISC-3 design-basis functional capability.

The evaluations performed to satisfy 50.69(c)(1)(iv) that sufficient safety margins are maintained must address potential impacts from known degradation mechanisms on both active and passive functions of SSCs. The manner for addressing these potential impacts during categorization may be either qualitative or quantitative, and may rely on the maintenance of current programs that address these degradation mechanisms (e.g., microbiologically-induced corrosion, flow-assisted corrosion) and/or may incorporate existing risk-informed approaches (e.g., risk-informed in-service inspection).

d. Licensees are permitted to implement 10 CFR 50.69 for a subset of the plant systems and structures (i.e., partial implementation) and to phase in implementation over time. However, the implementation, including the categorization process, must address entire systems or structures; not selected components within a system or structure. This required scope ensures that all system functions associated with a plant system or structure are properly identified and evaluated when determining the safety significance of individual components within a system or structure and that the entire set of components that comprise a system or structure are considered and addressed.

   System boundaries must be well defined and consistent with the categorization process. For example, electrically powered components will normally interface with the electric power systems through an isolation breaker. An isolation breaker may be a portion of the component, or it may be a component in the electrical system. The inspector should confirm that the interface is clearly defined so that all SSCs are assigned to a system. For components modeled in a PRA, the PRA models should properly reflect the component boundaries and interfaces used in the categorization.

   Additional guidance on system boundary definitions is included in the NRC Regulatory Guidance 1.201 and in the industry guidance document, NEI 00-04.

e. The determination of safety significance of SSCs is to be performed as part of an integrated decision-making process that integrates both risk insights and traditional engineering insights. The insights and varied experience of IDP members are relied on to ensure that the result reflects a comprehensive and justifiable judgment.

   The IDP must be composed of experienced personnel who possess diverse knowledge and insights in plant design and operation, and who are capable in applying deterministic knowledge and risk insights in making SSC classifications. At least three members of the IDP should have a minimum of five years of experience at the plant, and there should be at least one member of the IDP who has worked on the modeling and updating of the plant-specific PRA for a minimum of three years. The IDP should be trained in the specific technical aspects and requirements related to the categorization process.

   The licensee (through the IDP) should document its decision criteria for categorizing SSCs as safety significant or low safety significant. Decisions of the IDP should be arrived at by consensus. If a resolution cannot be achieved concerning the safety significance of an SSC, then the SSC should be classified as safety significant.

f. Current regulatory requirements for special treatment of RISC-1 SSCs are unaffected by implementation of 50.69 and must be retained. The current treatment of RISC-2 SSCs which support the credit taken for the performance of design basis functions in the PRA should be maintained. However, PRAs may credit RISC-1 and RISC-2 SSCs to perform functions beyond their design basis.

   Because RISC-1 and RISC-2 SSCs are the safety significant SSCs and their performance as credited in the PRA is important to maintaining an acceptable level of plant risk, it is necessary to ensure these SSCs can perform as credited in the PRA. The requirements in 10 CFR 50.69(d)(1) do not extend special treatment requirements to RISC-1 and RISC-2 SSCs beyond design-basis functions, but the inspector should confirm that an evaluation of the SSCs ability to perform beyond design basis functions to the extent credited in the PRA has been completed and that treatment has been modified as appropriate.

   For example, if a relief valve is credited with the capability to relieve water (as opposed to its design condition of steam), the language in 10 CFR 50.69(d)(1) requires an evaluation to determine whether the component would be able to perform as assumed.

g. While RISC-3 SSCs may periodically fail, focus should be placed on those SSCs whose failure rate has substantively increased. This evaluation should be performance-based, meaning that inspectors should review conditions in which SSCs' design functions were challenged and, in those situations, determine whether licensee's alternate treatment being applied to RISC-3 SSCs had a notable adverse impact on the identified deficient condition. In these situations, inspectors should evaluate the adequacy of the licensee's existing alternate treatment for RISC-3 components.

   In situations where there were no conditions identified in which SSCs' design-related functions were challenged by RISC-3 SSC failures, select one to three RISC-3 SSCs for each of the selected systems, and verify that the licensee has implemented their established alternate treatment program for these RISC-3 SSCs or has maintained the original special treatment requirements.

   Other aspects to inspect in this area include ensuring that: 1) RISC-3 valves in the selected system are placed in their correct positions; 2) inspection and testing and the corrective action applied to RISC-3 SSCs are reasonable based on their low-risk; 3) extent of conditions review in cases where existing alternate treatment being applied to RISC-3 SSCs were found to be deficient and had a notable adverse impact on the function of the system; 4) components that can be categorized as either RISC-1 or RISC-3 are properly segregated in storage to minimize the potential for installation of a component categorized as RISC-3 into a RISC-1 application and 5) the licensee is identifying and scheduling deficient conditions, such as corrosion, missing fasteners, cracks, and degraded insulation, for repair.

   Additional guidance for inspection of alternate treatment of RISC-3 SSCs can be found in appendix A.

h. The regulation states that existing information in the quality assurance plan or in the UFSAR may need to be revised to reflect the changes in treatment that are made as a result of implementation of 10 CFR 50.69. Any revisions to these documents are to be submitted to the NRC in accordance with the existing requirements of 10 CFR 50.54(a)(2) and 50.71(e).

i. The inspector should review licensee event reports for the selected SSCs to verify the licensee is appropriately reporting equipment deficiencies and events consistent with the requirement of 10 CFR 50.69(g). For RISC-1 and -2 SSCs confirm the licensee is implementing reporting requirements not required by 10 CFR 50.69(g), i.e., 10 CFR Part 21, 50.72, and 50.73 reporting requirements. Only RISC-3 and RISC-4 SSCs are excluded from 10 CFR part 21, 50.72, and 50.73 reporting requirements.

Title 10 CFR 50.69(g) provides a new reporting requirement applicable to events or conditions that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function. Most events involving these SSCs will meet existing 10 CFR 50.72 and 73 reporting criteria. However, it is possible for events and conditions to arise that impact whether RISC-1 or RISC-2 SSCs would perform beyond design-basis functions consistent with the performance capability credited in the categorization process.

03.03 Problem Identification and Resolution

For the selected systems, inspectors should review licensee corrective actions in accordance with the guidance of Inspection Procedure 71152, "Problem Identification and Resolution."

03.04 Review of Licensee's Feedback and Process Adjustments

a. The licensee must update the categorization or treatment processes in a timely manner without waiting for the two refueling outage schedule specified in 10 CFR 50.69(e)(1) if plant changes, operational practices, or operational experience would result in a significant adverse impact on plant safety or public health and safety.

The regulation emphasizes the importance of applying operating experience in maintaining plant safety. In particular, 10 CFR 50.69(e)(1) requires the feedback of plant operational experience in addition to the requirements to feedback performance data, plant changes, operational changes, and industry experience. This plant operational information may be obtained from the corrective action program and processes, as well as other sources.

In addition to the periodic updating of the quantitative reliability information, the feedback of plant operational experience is intended to include qualitative information on the performance of plant SSCs obtained through the corrective action program and processes as well as from applicable vendor recommendations and operational experience. For example, lessons learned from operational experience might be described in NRC information notices or implemented in response to NRC bulletins or generic letters. All of this information is needed to update both the PRA and SSC categorization and treatment processes.

b. In accordance with10 CFR 50.69(e)(2), plant documentation should indicate that the licensee is monitoring the performance of RISC-1 and RISC-2 SSCs and is making adjustments as necessary to either the categorization or treatment to ensure that the categorization basis remain valid.

To meet this requirement, the licensee must monitor all unavailability situations and functional failures so they can determine when adjustments to the categorization or treatment processes are needed. The licensee will also need to monitor SSCs that are credited in the PRA for performing beyond design-basis functions (if applicable) that are not necessarily included in the scope of an existing maintenance rule program

c. In accordance with 10 CFR 50.69(e)(3), plant documentation should indicate that the licensee is evaluating data collected in 10 CFR 50.69(d)(2)(i) for RISC-3 SSCs to determine whether any adverse changes in performance exist such that the SSC unreliability values for PRA modeled components approach or exceed the values used in the evaluations conducted to satisfy 10 CFR 50.69(c)(1)(iv) and that the licensee is making adjustments as necessary to the categorization or treatment processes to ensure that the categorization process and results remain valid.

   The evaluation of performance data collected through 10 CFR 50.69(d)(2)(i) is to determine whether there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations conducted to meet 10 CFR 50.69(c)(iv) and to make adjustments as necessary to either the categorization or treatment processes so the categorization process and results are maintained valid. Based on the review of this information, if SSC reliability degrades so as not to support the categorization process assumptions, the licensee must adjust the treatment to improve SSC reliability or make appropriate changes to the categorization of SSCs.

03.05 Training

Inspectors should review the training resources identified in the “10 CFR 50.69 Inspection Guidance and Resources” document. Additionally, just-in-time and specific topic training can be provided by NRR/DRA at a region’s request.

03.06 <u>Feedback and Lessons Learned</u>

The inspection team lead should provide any feedback, lessons learned, or best practices through the implementation of this procedure to the program office. The intent of this feedback process is to assist the program office in evaluating inspection efficiency and identifying appropriate guidance revisions or inspector training needs.

37060-04 RESOURCE ESTIMATE

A team leader and a three-member inspection team should conduct the inspection activities prescribed in this IP. Team composition should consider members with the following background: (1) mechanical, (2) electrical, (3) a Senior Reactor Analyst (SRA), and (4) current resident or senior resident inspector for the site being inspected. The team need not have an SRA as a member if another team member has had similar training and experience, is a qualified inspector, and the regional SRAs are available for consultation. If the resident or senior resident inspector are not available, one member of the team should have a reactor operational background relevant to the site being inspected for understanding the overall systems interaction and risk to safety.

Inspection of all elements in this procedure is expected to require between 250 – 310 hours (280 hours nominal) of direct inspection onsite or from the regional office.

If performed in phases, completion of each section is expected to require:

Section 02.01, Review of Licensee Programs and Procedures	90 hours
Section 02.02, Review of Licensee’s 50.69 Program Implementation	155 hours
Section 02.03, Problem Identification and Resolution	5 hours
Section 02.04, Review of Licensee’s feedback and process adjustments	30 hours

37060-05 PROCEDURE COMPLETION

This inspection procedure is intended to be completed in phases or in its entirety. Sections 02.01 and 02.02 should be performed after the licensee has categorized a sufficient number of systems such that an appropriate risk-informed selection of three to five systems can be selected for inspection. Section 02.04 should be performed following licensee implementation of the feedback and process adjustment programs. This would be typically following the second refueling outage after system categorization. The systems selected for inspection during the performance of section 02.02 need not be the same selected for inspection during performance of section 02.04. Section 02.03 should be completed each time the licensee is inspected.

Satisfactory completion of this inspection procedure is accomplished through completion of individual sections of this IP or through a combination of multiple sections. Licensee notification of this inspection should include which sections of the inspection procedure will be performed. Regions do not have to re-inspect licensee’s programs or implementation of their programs previously found to be acceptable.

37060-06 REFERENCES

NOTE: Some references contain hyperlinks to the specific document. These hyperlinks should be used with caution (the linked document should be verified to be the current version prior to use).

10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission, Washington, DC.

69 FR 68008, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” *Federal Register*, Volume 69, Number 224, p. 68047, Washington, DC, November 22, 2004.

COM-106, “Technical Assistance Request (TAR) Process” August 24, 2020. (ML19228A001) (non-public)

NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” Nuclear Energy Institute, Washington, DC, July 31, 2005. (ML052900163)

RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” U.S. Nuclear Regulatory Commission, Washington, DC. (ML090410014)

RG 1.201 “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to Their Safety Significance,” U.S. Nuclear Regulatory Commission, Washington, DC. (ML061090627)

SECY-04-0109, “Final Rulemaking To Add New Section 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,’” U.S. Nuclear Regulatory Commission, Washington, DC, June 30, 2004.

END

Appendices:

Appendix A: “Additional Guidance for the Alternate Treatment of RISK-3 SSCs”
Appendix B: “Guidance for 10 CFR 50.69 Reportability”

Attachment 1: Revision History for IP 37060

Appendix A: Additional Guidance for the Alternate Treatment of RISC-3 SSCs

In the *Federal Register* notice dated November 22, 2004 (69 FR 68008) for Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," of Title 10 of the *Code of Federal Regulations* (10 CFR 50.69), the Commission provided guidance for the alternate treatment of structures, systems, and components (SSCs) classified as Risk-Informed Safety Class (RISC)-3, which are safety-related SSCs that perform low safety significant functions. This appendix summarizes specific aspects of the guidance provided by the Commission for use by U.S. Nuclear Regulatory Commission (NRC) inspectors in evaluating the alternate treatment applied by licensees to RISC-3 SSCs as part of their 10 CFR 50.69 programs.

For RISC-3 SSCs, the NRC regulations in 10 CFR 50.69(b)(v), in part, remove the inservice testing requirements in 10 CFR 50.55a(f); and the inservice inspection, and repair and replacement (with the exception of fracture toughness), requirements for American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) Class 2 and Class 3 SSCs in 10 CFR 50.55a(g). Fracture toughness is an important material property that prevents premature failure of an SSC at abrupt geometry changes, or at small undetected flaws. Adequate fracture toughness of SSCs is necessary to prevent common cause failures due to design-basis events, such as earthquakes. Therefore, the Commission retained the fracture toughness requirements within the scope of repair and replacement of ASME components categorized as RISC-3 SSCs. See 69 FR 68008, 68013.

The Commission noted that the NRC regulations in 10 CFR 50.69(d)(2) explicitly require that the treatment of RISC-3 SSCs to be consistent with the categorization process because the treatment practices for plant SSCs must support the capability credited in the categorization process for there to be reasonable confidence that any increase in risk remains small. The use of voluntary consensus standards is one effective means to establish treatment requirements for RISC-3 SSCs. However, exercising a pump or valve may not be sufficient to ensure with reasonable confidence its design-basis capability. In Commission paper SECY-00-0194, "Risk-Informing Special Treatment Requirements," dated September 7, 2000, the NRC noted that a wide variation existed in industrial practices. Therefore, certain industrial practices alone may not be sufficient to satisfy the treatment requirements for RISC-3 SSCs in 10 CFR 50.69. As a result, the Commission required in 10 CFR 50.69 that the treatment of RISC-3 SSCs must be consistent with the categorization process. One way to achieve this consistency could be the application of consensus standards. However, licensees must recognize that the application of such standards must meet the 10 CFR 50.69(d)(2) requirements to be acceptable. The determination of consistency between treatment and categorization (e.g., assumed reliability levels) also includes consideration of applicable operational experience. See 69 FR 68008, 68013, 68041-68042.

Although a specific list of design control attributes is not included in 10 CFR 50.69 for RISC-3 SSCs, the NRC regulations in 10 CFR 50.69 require licensees to ensure with reasonable confidence that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. The Commission stated that 10 CFR 50.69 does not change design-basis functional requirements, and 10 CFR 50.59 remains applicable to all changes to non-special treatment aspects of RISC-3 SSCs. Under 10 CFR 50.69, RISC-3 SSCs are excluded from special treatment requirements for qualification methods for environmental conditions and effects, and seismic conditions. However, the requirements of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," continue to apply to SSCs important to safety within the scope of 10 CFR 50.69. As a result, RISC-3 SSCs continue to be required to be capable of performing their safety-related functions under applicable environmental conditions and effects and seismic conditions. As allowed by 10 CFR 50.69, licensees may provide a lower level of confidence (referred to as reasonable confidence in 10 CFR 50.69) than provided for RISC-1 SSCs when demonstrating the design-basis capability of RISC-3 SSCs to perform their safety-related functions under applicable environmental conditions and effects and seismic conditions. See 69 FR 68008, 68013-68014.

The Commission stated that 10 CFR 50.69 does not alter the existing seismic design requirements for RISC-3 SSCs in any plant's design basis. In meeting 10 CFR 50.69, the licensee must have adequate technical bases to conclude that RISC-3 SSCs will perform their safety-related functions under seismic design-basis conditions, which includes the number and magnitude of earthquake events specified for the SSC design. While the use of earthquake experience data is not prohibited by the regulation, the Commission noted that it may be difficult for a licensee to show that experience data alone will satisfy the applicable design requirements of 10 CFR Part 100. Under 10 CFR 50.69, the Commission stated that RISC-3 SSCs continue to be required to function under design-basis seismic conditions (such as design load combinations of normal and accident conditions with earthquake motions), albeit at a lower level of confidence than for RISC-1 SSCs, but would not be required to be qualified by testing or specific engineering methods in accordance with the requirements stated in 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." See 69 FR 68008, 68014, and 68041.

The NRC regulations in 10 CFR 50.69(d)(2) impose requirements that are intended to maintain RISC-3 SSC design-basis capability. Although individual RISC-3 SSCs are not significant contributors to plant safety, the Commission stated that RISC-3 SSCs perform functions necessary to respond to certain design-basis events of the facility. Thus, collectively, RISC-3 SSCs can be safety significant and as such, it is important to maintain their design-basis functional capability. In order to meet the regulatory requirements, licensees will need to obtain data or information sufficient to make a technical judgment that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions, and to restore equipment performance consistent with corrective action requirements included in the regulation. See 69 FR 68008, 68019-68020.

The Commission stated that a licensee is required to provide a "reasonable confidence" level with regard to maintaining the capability of RISC-3 safety-related functions. Although 10 CFR 50.69(b)(1) removes the environmental qualification requirements of 10 CFR 50.49 for RISC-3 SSCs, it does not eliminate the requirements in 10 CFR Part 50, Appendix A, that electric equipment important to safety be capable of performing their intended functions under the applicable environmental conditions. For example, General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," of 10 CFR Part 50, Appendix A, requires that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions and effects associated with normal operation, maintenance, testing, and postulated accidents.

In its Memorandum and Order CLI-80-21, dated May 23, 1980 (11 NRC 707 also located at ADAMS Accession No. ML20049A257), the Commission ordered that NUREG-0588 (initially issued in November 1979), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," and the Guidelines for Evaluating Qualification of Class 1E Electrical Equipment in Operating Reactors provided in a memorandum from H.R. Denton, Director, NRC Office of Nuclear Reactor Regulation, to V. Stello, Director, NRC Office of Inspection and Enforcement, dated November 13, 1979 (referred to as the DOR

Guidelines), must be satisfied to meet the requirements of 10 CFR Part 50, Appendix A, GDC 4. As discussed in NUREG-0588, the licensee must address environmental conditions such as temperature, pressure, humidity, chemical effects, radiation, and submergence; and environmental effects such as aging and synergisms, when demonstrating the design-basis capability of electrical components.

As allowed by 10 CFR 50.69, licensees may provide a lower level of confidence when demonstrating the design-basis capability of RISC-3 SSCs. A licensee implementing 10 CFR 50.69 for RISC-3 electrical components must consider operating life (aging) and combinations of operating life parameters (synergistic effects) including applicable environmental and operational service conditions when defining the alternative treatments for RISC-3 SSCs that ensure with reasonable confidence that their design-basis functionality is maintained throughout their service life. This is particularly important if the equipment contains materials which are known to be susceptible to significant aging mechanisms (e.g., degradation due to thermal, radiation or cyclic/wear) or synergistic effects that could impair the ability of the equipment to perform its design-basis function. The licensee may consider whether an electrical component was qualified in accordance with 10 CFR 50.49 such that the primary consideration when implementing 10 CFR 50.69 is degradation related to aging and synergistic effects to determine the end of service life of the RISC-3 electrical component with reasonable confidence. See 69 FR 68008, 68013-68014, and 68040-68041.

In establishing treatment for RISC-3 SSCs, the Commission stated that the licensee is responsible for addressing applicable vendor recommendations and operational experience such that the treatment established for RISC-3 SSCs provides reasonable confidence of design-basis capability. The treatment applied to RISC-3 SSCs must also support the assumptions used in justifying removal of requirements applicable to those SSCs. For example, where a licensee intends, as part of implementing 10 CFR 50.69, to eliminate leakage testing required in 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” for containment isolation valves, the treatment applied to those valves must support the assumption that they are capable of closing under design-basis conditions. See 69 FR 68008, 68041.

Some public comments on the proposed 50.69 rule suggested that a reference to general industrial practices would be sufficient to satisfy the requirements for the treatment for RISC-3 SSCs. However, as described in NUREG/CR-6752, “A Comparative Analysis of Special Treatment Requirements for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants,” significant variation exists in the application of industrial practices at nuclear power plants. The Commission stated that a simple reference to these practices does not provide a basis to satisfy the rule’s requirements. The licensee must establish treatment that provides reasonable confidence that SSCs will perform their safety-related functions under design-basis conditions and is consistent with the assumptions in the categorization basis (e.g., reliability levels, if applicable). The licensee must establish treatment that provides this level of reliability or use consensus standards that provide a proven level of reliability based on experience. In using consensus standards, the licensee must note that combining or omitting provisions of standards might result in ineffective implementation of 10 CFR 50.69 by causing RISC-3 SSCs to be incapable of performing their design-basis safety functions. The NRC considers the ASME Code Cases endorsed in 10 CFR 50.55a and listed in Regulatory Guides (RGs) 1.84, 1.147, and 1.192 to be one acceptable method of establishing treatment of RISC-3 SSCs, where applicable, in that those code cases adjust treatment based on the safety significance of the components. See 69 FR 68008, 68041-68042.

The Commission stated that the rule language 10 CFR 50.69(d)(2)(i) means that the licensee must implement periodic testing or inspection sufficient to provide reasonable confidence that RISC-3 pumps and valves will be capable of performing their safety-related functions under design-basis conditions. To determine that the pump or valve will remain capable of performing its safety-related function, the licensee will need to obtain sufficient operational information or performance data to provide with reasonable confidence that the RISC-3 pumps and valves will be capable of performing their safety-related functions if called upon to function under operational or design-basis conditions over the interval between periodic testing or inspections. In addition, the operational information and performance data must be sufficient to satisfy the requirements in 10 CFR 50.69 for identifying the need for corrective action and for feedback to the categorization and treatment processes. While 10 CFR 50.69 allows significant flexibility in verifying design-basis capability of RISC-3 SSCs, the licensee needs to consider the lessons learned regarding SSC performance in establishing the treatment for RISC-3 SSCs. For example, operating experience and research may not support an assumption that exercising a valve or pump will provide reasonable confidence of design-basis capability in that such exercising may not detect service-induced aging or degradation that could prevent the component from performing its design-basis functions in the future. The licensee may develop the type and frequency of tests or inspections for RISC-3 pumps and valves provided they are sufficient to conclude that the pump or valve will perform its safety-related function throughout the service life. The provisions for risk-informed inspection and testing in the applicable ASME Code Cases (as endorsed in 10 CFR 50.55a and listed in RGs 1.84, 1.147, and 1.192) would constitute one effective approach for satisfying the 10 CFR 50.69 requirements. Based on its discussion, the Commission stated that the inspection and testing requirement in 10 CFR 50.69(d)(2)(i) provides performance data for RISC-3 SSCs to determine if the reduction in treatment has adversely affected their design basis capability and to provide reasonable confidence that the SSC can perform its safety function throughout their service life. See 69 FR 68008, 68042.

The NRC regulations in 10 CFR 50.69(d)(2)(ii) require that conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design-basis conditions must be corrected in a timely manner. In the case of significant conditions adverse to quality, the regulation requires that measures be taken to preclude repetition. The Commission stated that significant conditions adverse to quality include common-cause concerns for multiple RISC-3 SSCs or concerns related to the validity of the categorization process or its results. For example, if measuring and test equipment is found to be in error or defective, the licensee will be responsible for determining the functionality of safety-related SSCs checked using that equipment to prevent the occurrence of common-cause problems that might invalidate the categorization basis or assumptions. Effective implementation of the corrective action process would include timely response to information from plant SSCs, overall plant operations, and industry generic activities that might reveal performance concerns for RISC-3 SSCs on both an individual and common cause basis. The Commission concluded that the corrective action requirement in 10 CFR 50.69(d)(2)(ii) is to provide reasonable confidence that RISC-3 safety-related functional capability is maintained and thereby avoid adverse impacts on the reliability and availability of multiple RISC-3 SSCs, which could reduce plant safety beyond the categorization process assumptions or results and invalidate the risk sensitivity results. See 69 FR 68008, 68043.

See NUREG-1482 (Revision 3, July 2020), “Guidelines for Inservice Testing at Nuclear Power Plants,” Appendix B, “Guidance for Treatment of Pumps, Valves, and Dynamic Restraints during Implementation of 10 CFR 50.69,” for additional information.

Appendix B: Guidance for 10 CFR 50.69 Reportability

The guidance presented in this appendix is for inspector use in determining reportability requirements for 10 CFR 50.69 categorized components.

<u>Applicable 50.69(g) reporting for beyond design basis impacts:</u>

A major focus of this requirement is to ensure that information is obtained for RISC-1 and RISC-2 SSCs regarding functions that are credited to mitigate beyond design basis accidents. The purpose for this reporting is to is enable the NRC to be aware of any situations impacting those functions found to be risk significant under 10 CFR 50.69. Therefore, reporting for credited beyond design basis functions is distinct from, and in addition to, any credited design basis functions for RISC-1 or RISC-2 SSCs.

<u>Situations that are appropriate for dual reporting requirements, including 50.69(g):</u>

Reporting under 10 CFR 50.72/50.73 for events involving design basis functions that are also associated with a RISC-1 or RISC-2 SSC does not eliminate the requirement for additional reporting under 10 CFR 50.69(g) for functions credited for beyond design basis mitigation associated with 10 CFR 50.69. Circumstances wherein event reporting for a RISC-1 or RISC-2 SSC involves a function that is credited for both a design basis function and a beyond design basis mitigation function requires dual reporting distinctly reflecting the reporting requirement that was satisfied for each function. For example, the loss of a single train safety function in the reactor core isolation cooling (RCIC) system at boiling water reactors (BWR) may be reportable under 10 CFR 50.72/50.73 (see RIS-01-14), and would also be reportable under 10 CFR 50.69(g) if the RCIC system is credited for a beyond design basis mitigation function as a RISC-1 or RISC-2 SSC.

<u>Proper documentation of 50.69(g) reporting in LERs:</u>

Current reporting requirements remain in place for RISC-1 and RISC-2 SSCs. In addition, per 10 CFR 50.69(g), LERs submitted for events involving RISC-1 or RISC-2 SSCs must report issues for both credited design basis functions and credited beyond deign basis mitigation functions under 10 CFR 50.73(b). LERs submitted per 10 CFR 50.73(b) for events involving only design basis functions associated with RISC-1 or RISC-2 SSCs should also provide the reasons why beyond design basis functions are not impacted or reportable under 10 CFR 50.69(g) requirements as a good practice.

<u>Reporting applicability based at the system level focused on fulfillment of the safety function rather than at the component level:</u>

The NRC regulations in 10 CFR 50.69(g) require reporting of events or conditions that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function. In the *Federal Register* dated November 22, 2004 (69 FR 68008, 68044), the Commission explained that 10 CFR 50.69(g) provides a new reporting requirement applicable to events or conditions that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function. The Commission noted that most events involving these SSCs will meet existing 10 CFR 50.72 and 50.73 reporting criteria. However, it is possible for events and conditions to arise that impact whether RISC-1 or RISC-2 SSCs would perform beyond design basis functions consistent with the performance capability credited in the categorization process. This reporting requirement is intended to capture these situations. The reporting requirement is contained in 10 CFR 50.69, rather than as a revision of 10 CFR 50.73, so that its applicability only to those facilities that have implemented 10 CFR 50.69 is clear. The existing reporting requirements in 10 CFR 50.72 and 50.73 are removed for RISC-3 and RISC-4 SSCs under 10 CFR 50.69(b)(vii) and (viii). Further, at 69 FR 68008, 68023, the Commission noted that although 10 CFR Part 21 and 10 CFR 50.55(e) (component defect) reporting will not be required for RISC-2 SSCs, 10 CFR 50.69(g) contains enhanced reporting requirements applicable to the loss of system function attributable to, inter alia, failure or lack of function of RISC-2 SSCs.

The Commission's statement regarding the intent of paragraph (g) also makes it clear that the Commission did not intend the reporting requirements contained in 10 CFR 50.69(g) to change or expand the reporting requirements in 10 CFR 50.72 and 73 applicable to design-basis functions. Therefore, the NRC staff interprets this Commission guidance to indicate that reporting of component failures for RISC-1 or RISC-2 SSCs where redundant equipment in the same system was available to perform the relevant safety significant function is not required. Loss of an entire RISC-1 or RISC-2 system to perform its credited design-basis safety functions is reportable under 10 CFR 50.72/50.73, as applicable, and reportable under 10 CFR 50.69(g) for the loss of credited beyond design basis functions, even if another separate system is available to perform the credited safety functions.

Attachment 1: Revision History for IP 37060

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	ML102700396 09/14/11 CN 11-016	Reviewed commitments and found none for 4 years. Developed new inspection procedure to provide inspection guidance on 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants."	Inspector training required prior to using this new inspection procedure	ML103081058
N/A	ML20192A322 07/28/20 CN 20-035	Feedback Form (FF) 37060-2299 incorporated to include references to ANO2-R&R-004 and RG 1.174. Periodic review completed per IMC0307A, no changes at this time other than to incorporate FF 37060-2299. Working group is being proposed to develop enhancements to this IP.	None	None FBF 37060-2299 ML20202A167
N/A	ML21273A303 11/29/21 CN 21-038	Complete rewrite to IMC 0040 format. Additional changes and guidance enhancements as a result of Lessons Learned Working Group following February 2020 Limerick inspection (Recommendation Memo ML21182A110) IMC 0040 format exception approved by IRIB branch chief.	None	ML21273A301
N/A	ML22075A287 10/18/22 CN 22-023	Minor revision to guidance contained in appendix A to clarify intent of inspector review of alternate treatment for RISC-3 SSCs	None	ML22224A211