NRC INSPECTION MANUAL IRIB

INSPECTION PROCEDURE 37060

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

PROGRAM APPLICABILITY: 2515

037060-01 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 and described in the plant safety evaluation and Updated Final Safety Analysis Report (UFSAR) and documented in the staff's Safety Evaluation Report (SER).

Comment [A1]: The SER may contain information that is not considered to be commitments required to be included in the licensee's programs and procedures.

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 and subjected to alternative treatments.

37060-02 INSPECTION REQUIREMENTS AND GUIDANCE

02.01 Review of the Licensee's Programs and Procedures. The inspector should review the licensee's programs and procedures to ensure that the procedures fully describe the categorization and treatment process (to the extent that alternative treatments have been implemented) for SSCs as described in its UFSAR and as required by 10 CFR 50.69. Specifically, inspectors should verify the following aspects of the licensee's programs and procedures:

a. The process for categorization (based on active functions, passive pressure boundary functions, and functions relied upon to respond to initiating events not modeled in the PRA) of SSCs into risk-informed safety class (RISC)-1, RISC-2, RISC-3 and RISC-4 categories~~SSCs~~ using probabilistic risk assessment (PRA) insights and by means other than PRA models described in their procedure(s) is consistent with the categorization process approved by NRC.

Before a licensee can implement the requirements in 10 CFR 50.69, the NRC must approve the categorization process. A licensee will submit an application for a license amendment under 10 CFR 50.90 that contains the information required by 10 CFR 50.69(b)(2). The NRC will approve a licensee's application of 10 CFR 50.69 by issuing a license amendment if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c)~~ by issuing a license amendment~~. This approval is necessary because of the importance of the PRA and categorization process to the successful implementation of the regulation. This review and approval of the categorization process is a one-time process approval. The approval is not restricted to a set of systems or structures and can be applied to any system or structure in the plant. In addition, the licensee is not required to return to the NRC for review of the categorization process 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.

The description of the categorization of SSCs into RISC-1, RISC-2, RISC-3, and RISC-4 categories~~SSCs~~ 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 internal and external initiating events~~seismic, fire, and other initiating events that are modeled in the PRA~~.

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

The regulation at 10 CFR 50.69(c)(1)(ii) requires the categorization process to determine the functional importance of SSCs, using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operational experience.

c. The procedure(s) describes how the licensee maintains defense-in-depth~~ and safety margins~~.

The regulation at 10 CFR 50.69(c)(1)(iii) requires the licensee to maintain defense-in-depth as part of the categorization process. The regulation at 10 CFR 50.69(c)(1)(iv) requires the licensee to consider the revised treatment applied to RISC-3 SSCs for its potential impact on risk. For example, the containment and its systems are important in the preservation of defense-in depth (in terms of both large early and large late releases). Inspectors should evaluate the licensee's defense-in-depth evaluations to confirm that they are properly implementing the approved process.

d. The procedure(s) describe the basis for the acceptability of the evaluations to be conducted to provide reasonable confidence that the licensee is maintaining sufficient safety margins.

The regulations at 10 CFR 50.69(c)(1)(iv) requires that the licensee's process includes 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 permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small. ~~Inspectors should confirm that applicable codes and standards are being applied, or that the licensee's implementation of the process provides confidence that safety analysis acceptance criteria in the licensing bases (e.g., FSAR, supporting analyses) are met.~~

e. The procedure(s) require evaluation of entire systems and structures, not just selected components within a system or structure.

The regulation at 10 CFR 50.69(c)(1)(v) requires licensees to 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 safety 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 P & ID or a single line diagram) that comprise a system or structure.

f. The procedure(s) require SSCs to be categorized 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.

Comment [A2]: safety margins are covered in the next section

Comment [A3]: Nothing in the 50.69 rule requires the use of codes and standards or the evaluation against safety analysis acceptance criteria

The regulation at 10 CFR 50.69(c)(2) requires an IDP to evaluate the risk insights and other traditional information; this panel must comprise expert, plant-knowledgeable members whose expertise includes PRA, safety analysis, plant operation, design engineering, and system engineering. 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 using deterministic knowledge and risk insights to categorize SSCs.

g. The procedure(s) describes treatment applied to RISC-1 (only for beyond design basis events) and RISC-2 SSCs to ensure that these SSCs will perform their functions consistent with the categorization ~~process~~ basis~~assumptions~~. The inspectors should confirm that the treatment for ~~all~~ RISC-1 (only for beyond design basis events) and RISC-2 SSCs is ~~described in procedures and is~~ being correctly applied.

The regulation at 10 CFR 50.69(d)(1) imposes requirements that are intended to ensure that RISC-1 and RISC-2 SSCs will perform their functions consistent with the categorization ~~process~~ basis~~assumptions~~. The regulations require the licensee or applicant to evaluate the treatment being applied to RISC-1 and RISC-2 SSCs to ensure it supports the key assumptions in the categorization ~~process~~ basis that relate to the assumed performance of these SSCs.

h. The procedure(s) describes alternate treatment applied to RISC-3 SSCs to ensure that these SSCs will continue to be capable of performing their safety-related functions under design-basis conditions.

The licensee or applicant 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 treatment of RISC-3 SSCs must be consistent with the categorization basis~~process~~. This means that the licensee or applicant must establish treatment ~~- simple reference to general industrial practices does not provide a basis to satisfy the intent of the regulatory requirements -~~ that provides reasonable confidence SSCs perform their safety-related functions under design basis conditions and is consistent with the assumptions in the categorization ~~process~~ basis (e.g., reliability levels, if assumed in the categorization basis). ~~The licensee or applicant must establish treatment that provides this level of reliability, for example, use consensus standards or NRC accepted guidance that provide a proven level of reliability based on experience. In using consensus standards, the licensee or applicant must note that combining or omitting provisions of standards might result in ineffective implementation of § 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 RGs 1.84, 1.147, and 1.192 to be one acceptable method of establishing~~

Comment [A4]: Nothing in the 50.69 rule requires this or hints that this is required to satisfy the intent.

~~treatment of RISC-3 SSCs, where applicable, in that those code cases adjust treatment based on the safety significance of the components.~~

Inspection and testing, and corrective action shall be provided for RISC-3 SSCs. Periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions. 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. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

The inspection and testing requirement in § 50.69(d)(2)(i) is to provide sufficient 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. The corrective action requirement in § 50.69(d)(2)(ii) is to address SSC failures and provide reasonable confidence in avoiding future problems. These requirements are necessary 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~~ basis assumptions or results and invalidate the risk sensitivity results.

i. The procedure(s) require feedback and process adjustments.

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 if those factors were credited in the categorization ~~process~~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~~ basis or results.

Consequently, 10 CFR 50.69(e) contains requirements for updating the categorization and treatment processes when conditions warrant to ensure that categories assigned to SSCs continues to reflect the performance of the SSCs and the as-built, as-operated facility. Specifically, the regulation requires licensees to review the changes to the plant, operational practices, and applicable plant and industry operational experience and to update, as appropriate, the PRA and SSC categorization. Licensees must perform the review in a timely manner but no longer than once every two refueling outages. In addition, licensees must obtain sufficient information on SSC performance to verify that the categorization ~~process~~ basis ~~and its results~~ remains valid.

Comment [A5]: Nothing in the 50.69 rule requires or suggests the use of codes and standards. This should be left to the licensee to determine, based on reasonable confidence, what treatment is needed. In addition, reliability may not have been a factor in the categorization basis for a particular component and therefore an adverse change in the reliability of that component would have no impact on its risk significance. An example is a local indicating device or a normally closed 3/8" vent valve whose failures would have no risk significant impact even during a design basis event.

j. The procedure(s) requires program documentation, change control, and maintenance of records.

The regulation at 10 CFR 50.69(f) specifies requirements for the documentation of the program, the control of plant programs and procedures, and the maintenance of records. In particular, 10 CFR 50.69(f)(1) requires the licensee or applicant to document the basis for its categorization of any SSC before removing any special treatment requirements listed in 10 CFR 50.69(b)(1) from these SSCs. The regulation at 10 CFR 50.69(f)(2) requires licensees and applicants to update their final safety analysis report. The regulation at 10 CFR 50.69(f)(3) specifies that for initial implementation of the regulation, changes to the FSAR for implementation of this regulation need not include a supporting § 50.59 evaluation of changes directly related to implementation. Future changes to the treatment processes and procedures for § 50.69 implementation may be made, provided the requirements of the regulation and § 50.59 continue to be met. While the licensee is to update its programs to reflect implementation of § 50.69, the Commission concluded that no additional review under § 50.59 is necessary for such changes to these parts of the FSAR that might occur. Title 10 CFR 50.69(f)(4) section specifies that for initial implementation of this regulation, changes to the quality assurance plan directly related to implementation of this regulation need not be considered a reduction in commitment for the purposes of § 50.54(a). Future changes to the treatment processes and procedures for § 50.69 implementation may also be made, provided the requirements of the regulation and § 50.54(a) continue to be met. While the licensee is to update its programs to reflect implementation of § 50.69, the Commission concluded that no additional NRC staff review under § 50.54(a) is necessary for changes to these parts of the QA plan.

k. The procedure(s) contains requirements for reporting.

The regulation at 10 CFR 50.69(g) requires the licensee to submit a Licensee Event Report (LER) under 10 CFR 50.73(b) for any event or condition that would have prevented RISC-1 or RISC-2 SSCs from performing a safety-significant function. The licensee’s plant procedures should ensure that the LER process and the corrective action program under 10 CFR 50.69 conform to this regulation. For RISC-1 and RISC-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 -4 SSCs are exempt from 10 CFR part 21, 50.72, and 50.73 reporting requirements.
02.02 Review of the Licensee’s 10 CFR 50.69 Program Implementation. The inspector should sample one to five systems that the licensee evaluated using its approved 10 CFR 50.69 categorization process. The inspector should verify the implementation of the 10 CFR 50.69 as follows:

a. SSCs were properly categorized

The inspector should confirm that the licensee properly categorized key SSCs that can affect the system safety functions. The inspector should sample the basis for categorization of several SSCs (particularly RISC-3 SSCs). The licensee should have adequately documented the basis for its categorization. The inspector should forward, via regional management, concerns to the Inspection Program Branch (IRIB) in NRR if he or she cannot resolve any aspects of the licensee's categorization results during the inspection. For the sampled 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.

The cornerstone of 10 CFR 50.69 is the establishment of a robust, risk-informed categorization process that provides high confidence that the safety significance of SSCs is correctly determined considering all relevant information. The process is structured to ensure that all relevant information pertaining to SSC safety significance is considered by a panel that has expertise and capabilities for making a sound decision regarding the SSC's categorization, and that the assembled information is considered in a manner that ensures the Commission's criteria for risk-informed applications are satisfied. This process enables SSCs to be placed in the correct RISC category so that the appropriate treatment requirements will be applied commensurate with the SSC's safety significance.

b. Plant-specific PRA models of severe accident scenarios used are maintained.

The regulation at 10 CFR 50.69(c)(1)(i) requires the PRA to be of sufficient quality and level of detail to support the categorization process.

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 must 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 configuration and 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.

c. The licensee has properly implemented their integrated, systematic categorization process.

Comment [A6]: This appears to be a requirement in the future (after NRC approval of the licensee's categorization process) which is not required by 50.69

For the sampled SSCs, the inspectors should confirm that the licensee has properly implemented their integrated, systematic process for determining the functional importance of the SSCs.

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 integrated decision-making panel (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.

d. Defense-in-depth and safety margin were maintained

For the sampled SSCs, the inspector should confirm that defense-in-depth is maintained where SSCs are categorized as RISC-3. 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. ~~RISC-3 SSCs are considered degraded if it is not fully capable of performing its intended safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life.~~

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.

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).

e. RISC-3 SSC categorization evaluations were properly performed

For the sampled 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 ~~all~~ PRA-modeled components categorized as

Comment [A7]: See use of phrase "significantly degraded' in the last paragraph in this section "d".

Comment [A8]: Defense in depth deals with preservation of the basic safety functions (reactivity control, core cooling, heat sink, RCS inventory, and containment barrier) and not with individual components.

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 correct (i.e., cumulative) results are being compared to the appropriate quantitative guidelines.

The NRC recognizes that the reliability of RISC-3 SSCs could potentially decrease due to the reduction in treatment applied to these SSCs as a result of 10 CFR 50.69 implementation.

The NRC also recognized that 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 be consistent with (i.e., maintain the validity of) the categorization ~~process~~basis; (4) feedback requirements of 10 CFR 50.69(e) to maintain the validity of the categorization ~~process~~basis; and (5) the high-level RISC-3 requirements designed to maintain, with reasonable confidence, 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).

f. Entire systems and structures were evaluated

For the sampled SSCs, the inspector should confirm that the licensee performed evaluations to categorize SSCs for entire systems and structures, not just for selected components within a system or structure.

Licensee is allowed 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 safety functions associated with a 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 (to a reasonable level of detail, e.g.; all SSCS depicted on a P & ID or a single line diagram) ~~-~~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.

g. Staffing of expert panel met requirements

The inspector should confirm that the licensee's IDP panel was staffed with members whose joint expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.
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 final 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 the use of deterministic knowledge and risk insights in making SSC classifications. At least three members of the IDP should have a minimum of 5 years 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 3 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.

h. RISC-1 and RISC-2 SSC treatment is consistent with credit taken for the SSCs in the licensee's PRA model

For the sampled SSCs, the inspector should confirm that the licensee evaluated the treatment applied to RISC-1 and RISC-2 SSCs to support the credit taken for these SSCs in its PRA model.
~~The regulations in 10 CFR 50.69 maintain C~~Current regulatory requirements for special treatment of RISC-1 and current treatment of RISC-2 SSCs are maintainedwhich will support the credit taken for the performance of design basis functions in the

PRA. However, PRAs may credit RISC-1 and RISC-2 SSCs to perform functions beyond their design basis.

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. 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 a key and necessary part of 10 CFR 50.69 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.

i. RISC-3 SSC alternate treatment

For the sampled SSCs, the inspector should confirm that the 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. This evaluation should be performance-based, meaning that inspectors should review conditions in which SSCs' system level design~~-related~~ functions were challenged and in those situations, determine whether licensee's alternate treatment being applied to RISC-3 SSCs had a notable adverse ~~impact~~ role in ~~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 sampled 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. Document the results of this inspection activity in an inspection report scope section.

Other aspects to inspect in this area include ensuring that: 1) RISC-3 valves in the sampled system ~~categorized as RISC-3~~ 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 safety significant function(s) ~~design~~ 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, commensurate with providing reasonable confidence that these conditions would not prevent the SSC from performing its safety related function under design basis conditions.~~.~~

Additional guidance is provided below in the area of alternate treatment for RISC- 3 SSCs:

The Agency noted that 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.

Title 10 CFR 50.69(d)(2) explicitly requires that the treatment of SSCs to be consistent with the categorization ~~process~~ basis because the treatment practices for plant SSCs must support the capability credited in the categorization ~~process~~ basis for there to be reasonable confidence that any increase in risk remains small.

The Agency determined that the use of voluntary consensus standards as one effective means to establish treatment requirements for RISC-3 SSCs. However, exercising a pump or valve was evaluated as not sufficient to ensure with reasonable confidence its design-basis capability. In Commission paper SECY-00-0194, the NRC noted that a wide variation existed in industrial practices. Therefore, certain industrial practices may not be sufficient to satisfy the treatment requirements for RISC-3 SSCs in 10 CFR 50.69. As a result, the Agency clarified Title 10 CFR 50.69 to indicate that the treatment of RISC-3 SSCs must be consistent with the categorization ~~process~~basis. 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.

The Agency determined that a specific list of design control attributes would not be included in 10 CFR 50.69 for RISC-3 SSCs. The regulation requires licensees to ensure with reasonable confidence that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Title 10 CFR 50.69 is not changing 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 will be exempt from special treatment requirements for qualification methods for environmental conditions and effects, and seismic conditions. Nevertheless, 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, albeit at a lower level of confidence as compared to RISC-1 SSCs. ~~Therefore, a licensee implementing 10 CFR 50.69 must consider operating life (aging) and~~

Comment [A9]: Compare this statement with the one in 02.01.h

~~combinations of operating life parameters (synergistic effects) in the design of RISC-3 electrical equipment. This is particularly important if the equipment contains materials which are known to be susceptible to significant degradation due to thermal, radiation, and/or wear (cyclic) aging including any known synergistic effects that could impair the ability of the equipment to meet its design-basis function.~~

To address a concern with~~that~~ the absence of requirements to consider common cause issues for RISC-3 SSCs in the regulation, the Agency specified in 10 CFR 50.69(d)(2)(ii) that, for significant conditions adverse to quality associated with RISC-3 SSCs, measures shall be taken to provide reasonable confidence that the cause of the condition is determined and corrective action is taken to preclude repetition.

This regulation 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 Agency 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.

Section 50.69(d)(2) imposes requirements that are intended to maintain RISC-3 SSC design-basis capability. Although individual RISC-3 SSCs are not significant contributors to plant safety, they do 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.

Licensee is required to provide a "reasonable confidence" level with regard to maintaining the capability of RISC-3 safety-related functions. Although 50.69(b)(1) removes for RISC-3 SSCs the environmental qualification requirements of 10 CFR 50.49, 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, GDC 4 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. To satisfy the provisions of GDC 4, the licensee must address environmental conditions such

Comment [A10]: Nothing in the 50.69 rule requires that these attributes be required, as stated earlier in the paragraph.

Comment [A11]: This paragraph should be moved elsewhere.

as temperature, pressure, humidity, chemical effects, radiation, and submergence; and environmental effects such as aging and synergisms.

Under 10 CFR 50.69, RISC-3 SSCs would 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, 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.

In establishing treatment for RISC-3 SSCs, 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 for containment isolation valves, the treatment applied to those valves must support the assumption that they are capable of closing under design-basis conditions.

In response to a concern that general industrial practices would be sufficient to satisfy the RISC-3 SSC treatment requirements, the Agency stated that an NRC study had found that significant variation exists in the application of industrial practices at nuclear power plants. Therefore, the Agency stated that a simple reference to these practices does not provide a basis to satisfy the regulatory 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 ~~process~~ 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 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.~~

Comment [A12]: Nothing in the 50.69 rule requires or implies this.

The statement in 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. In some cases, a licensee implementing 10 CFR 50.69 might apply more rigorous test methods than previously applied to satisfy the ASME Code inservice testing provisions because 10 CFR 50.69 does not specify restrictive time limits on test intervals that were provided in the ASME Code. 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 over the last 20 years regarding SSC performance in establishing the treatment for RISC-3 SSCs. For example, operating experience and research does not support an assumption that exercising a valve or pump will provide reasonable confidence of design-basis capability in that such exercising will 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 incorporated in 10 CFR 50.55a) would constitute one effective approach for satisfying the 10 CFR 50.69 requirements.
Title 10 CFR 50.69(d)(2)(ii) requires 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. Significant conditions adverse to quality include common-cause concerns for multiple RISC-3 SSCs or concerns related to the validity of the categorization ~~process~~ basis 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 if that might invalidate the categorization ~~process~~ basis or assumptions ~~and results~~. 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.

j. Feedback and process adjustments

For the sampled SSCs, the inspector should confirm that the licensee is implementing the requirements for feedback and process adjustment in 10 CFR 50.69(e). In accordance with 10 CFR 50.69(e)(1), plant documentation should indicate that the licensee is reviewing changes to the plant, operational practices, and applicable industry operational experience at least every other refueling outage and is making updates to the PRA and SSC categorization as appropriate. In accordance with 10 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 ~~processes~~ to ensure that the categorization ~~process~~ basis and results remain valid. 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~~ basis and results remain valid. As part of this evaluation, the inspector should 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.

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 feed back 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.

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. 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.

Title 10 CFR 50.69(e)(2) requires the licensee to monitor the performance of RISC-1 and RISC-2 SSCs, and make adjustments as necessary to either the categorization or treatment ~~processes~~ so the categorization ~~process~~ basis and results are maintained 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.

Title 10 CFR 50.69(e)(3) requires the licensee to consider the performance data collected in 10 CFR 50.69(d)(2)(i) for RISC-3 SSCs 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~~ basis and results are maintained valid. Based on the review of this information, if SSC reliability degrades so as not to support the categorization ~~process~~ basis assumptions, the licensee must adjust the treatment to improve SSC reliability or make appropriate changes to the categorization of SSCs.

k. Documentation

For the sampled SSCs, the inspector should confirm that the licensee is implementing the requirements for program documentation, change control, and records in 10 CFR 50.69(f). 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).

l. Reporting

For the sampled SSCs, the inspector should confirm that the licensee is implementing the reporting requirements in 10 CFR 50.69(g). If necessary, the inspector should review licensee event reports for additional sampled SSCs to determine the implementation of this requirement. 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 -4 SSCs are exempt 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~~basis.
02.03 General Guidance. The NRC has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to public health and safety, thereby providing reasonable assurance of adequate

protection to public health and safety. The NRC mainly uses a "deterministic" approach as the basis for the current body of its regulations and their implementation.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacturing, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design-basis events (DBEs). The deterministic approach contains implied elements of probability (qualitative risk considerations) from the selection of accidents to be analyzed (e.g., reactor vessel rupture is considered too improbable for inclusion) to the system level requirements for emergency core cooling (e.g., safety train redundancy and protection against single failure). The deterministic approach then requires that the licensed facility include safety systems capable of preventing and/or mitigating the consequences of those DBEs to protect public health and safety. Those SSCs necessary to defend against the DBEs are defined as "safety related," and these SSCs are the subject of many regulatory requirements designed to ensure that they are of high quality and high reliability and that they have the capability to perform during postulated design-basis conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety related," "important to safety," or "basic component." The regulations define the terms "safety related" and "basic component," whereas "important to safety," which is used principally in the general design criteria of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," is not explicitly defined.

These prescriptive requirements as to how licensees are to treat SSCs, especially those that are defined as "safety related," are referred to as "special treatment requirements." The NRC developed these requirements to provide greater assurance that these SSCs would perform their functions under particular conditions (e.g., seismic events or harsh environments) with high quality and reliability for as long as they are part of the plant. These requirements include particular examination techniques, testing strategies, documentation requirements, personnel qualification requirements, independent oversight, etc. In many instances, the NRC developed these "special treatment" requirements as a means to gain assurance when more direct measures (e.g., testing under design basis conditions or routine operation) could not show that SSCs were functionally capable.

Special treatment requirements are imposed on nuclear reactor applicants and licensees through numerous regulations that have been issued since the 1960s. These requirements specify different scopes of equipment for different special treatment requirements depending on the specific regulatory concern, but they are derived from consideration of the deterministic DBEs. Treatment for an SSC, as a general term, refers to activities, processes, and/or controls that the licensee performs or uses in the design, installation, maintenance, and operation of SSCs as a means of:

(1) Specifying and procuring SSCs that satisfy performance requirements;

(2) Verifying over time that performance is maintained;

(3) Controlling activities that could impact performance; and

(4) Providing the assessment and feedback of results to adjust activities as needed to meet desired outcomes.

Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between “treatment” and “special treatment” is the degree to which the NRC specifies what the licensee must implement for particular SSCs or for particular conditions.

Section 50.69 represents an alternative set of requirements whereby a licensee or applicant may voluntarily categorize its SSCs consistent with the requirements in §50.69(c), remove the special treatment requirements in §50.69(b) for SSCs that are determined to be of low individual safety significance, and implement the alternative treatment requirements in §50.69(d). The regulatory requirements that have not been removed by §50.69(b) and the requirements specified in §50.69 continue to apply. The regulation contains requirements by which a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. To implement these requirements, the licensee must use a risk-informed categorization process to determine the safety significance of SSCs and to place the SSCs into one of four RISC categories. The licensee must ~~perform~~ implement an integrated decision-making process that uses both risk insights and traditional engineering insights to determine safety significance. The safety functions include both the design-basis functions (derived from the “safety-related” definition, which includes external events) and functions credited for severe accidents (including external events). Treatment for the SSCs must be applied as necessary to maintain functionality and reliability and is a function of the category into which the SSC is categorized. Finally, periodic assessment activities are conducted to make adjustments to the categorization and treatment processes as needed to ensure that SSCs continue to meet applicable requirements. The regulation contains requirements for obtaining prior NRC review and approval of the categorization process and for maintaining certain plant records and reports.

Although the intent of 10 CFR 50.69 is to ensure that the scope of the special treatment requirements imposed on SSCs is risk-informed, it is not to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet the current requirements governing design change, most notably 10 CFR 50.59, “Changes, Tests, and Experiments”). Instead, §50.69 should enable licensees and the staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., “reasonable confidence”) that these SSCs will satisfy functional requirements.

Before performing any phase of this procedure, the inspectors should develop an understanding of 10 CFR 50.69 and the licensee's 10 CFR 50.69 ~~special~~ alternative treatment program. If the inspectors cannot resolve any aspects of the licensee's interpretation of or implementation of 10 CFR 50.69 during the inspection, they should forward, via regional management, their concern to the Inspection Program Branch (IRIB) in the Office of Nuclear Reactor Regulation (NRR) to determine whether a resolution of their issue(s) should be resolved using the Task Interface Agreement (TIA).

02.04 Advance Preparation. 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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML042960073).

c. RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to Their Safety Significance" (ADAMS Accession No. ML061090627).

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

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. The inspector should consider other documents as background information when preparing for this inspection. For example, the NRC staff prepared a safety evaluation dated August 2001 on the request by the South Texas Project (STP) for exemption from certain special treatment requirements for safety-related SSCs with low safety significance. In addition, the American Society of Mechanical Engineers (ASME) has prepared guidance for the treatment of low-risk safety-related pumps and valves in Part 29, "Alternative Treatment Requirements for RISC-3 Pumps and Valves," of the ASME OM Standards and Guides for the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), although the NRC staff has not reviewed or accepted the ASME guidance at this time.

37060-03 RESOURCE ESTIMATE

A team leader and a three-member inspection team should conduct the inspection activities prescribed in this IP. Members of this team should have the following background: (1) Mechanical, (2) Electrical, and (3) a Senior Reactor Analyst. One member of the team should have a reactor operational background for understanding the overall systems interaction and risk to safety.

The time necessary to perform this IP is estimated at 40 to 80 hours per team member. This translates into about .45 FTE for each time this inspection is conducted (assuming a team leader with three inspectors; a week of prep; two weeks on site; followed by one week for inspection documentation). The inspection team should review the licensee's program and its implementation for adequacy after the licensee has categorized at least one system in accordance with 10 CFR 50.69. An inspection team should perform additional inspections to verify that the licensee continues to properly implement 10 CFR 50.69 based on (1) an event(s) that would indicate that the licensee's RISC-3 or RISC-4 components may be more safety-significant than assigned, (2) whether the alternative treatments applied to the components categorized as RISC-3 were not sufficient to preclude degrading the reliability of the components beyond assumed in the change-in-risk sensitivity studies, or (3) the number of safety-related systems categorized under 10 CFR 50.69 and the length of time since an inspection team last inspected this area.

37060-04 TRAINING

The training of NRC inspectors to evaluate the implementation of the regulation will be needed to ensure that inspectors can properly verify licensee's implementation of the 10 CFR 50.69 licensing amendment. The NRC Headquarters staff will evaluate the description of the categorization and treatment (if submitted with the license amendment) processes as part of the license amendment review and will document its findings in a safety evaluation report. The inspector will need to discuss the license amendment and the safety evaluation report with cognizant NRC Headquarters staff to ensure that the inspection of a nuclear power plant implementing 10 CFR 50.69 is consistent with the staff's assumptions and findings in reaching a conclusion on the license amendment application. The interaction between the NRC Headquarters staff and inspectors may involve meetings or telephone conferences to discuss the overall objectives and requirements in 10 CFR 50.69 and the plant-specific license amendment and the applicable assumptions and findings.

37060-05 IMPACT ON OTHER INSPECTION ACTIVITIES

Impact on the baseline inspection activities for plants adopting 10 CFR 50.69 will be evaluated at a later date by the Inspection Program Branch in the Office of NRR.

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).

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.

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.

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.

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. (ADAMS Accession No. 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. (ADAMS Accession No. ML061090627)

COM-106, "Control of Task Interface Agreements," U.S. Nuclear Regulatory Commission, Washington, DC, March 17, 2008. (ADAMS Accession No. ML073440014)

NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Nuclear Energy Institute, Washington, DC, July 31, 2005. (ADAMS Accession No. ML052900163)

NRC Safety Evaluation Report on STP Exemption Request, August 3, 2001 (ADAMS Accession Nos. ML011990368 and ML012040370).

WCAP-16308-NP-A, Revision 0, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program - Categorization Process - Wolf Creek Generating Station," August 2009 (ML092430186).

37060-07 COMPLETION STATUS

This inspection procedure shall be conducted to demonstrate that the licensee has satisfactorily implemented 10 CFR 50.69 requirements. This inspection can be performed in phases or in its entirety after NRR approves the 10 CFR 50.69 license submittal for the facility. Satisfactory completion of this inspection procedure is accomplished through performing all inspection requirements (performing the nominal number of samples where indicated (preferred) or the minimum number of samples in cases where nominal number of samples are not available) identified in sections 02.01 and 02.02 of this inspection procedure. It is acceptable for regions to perform portions of this inspection procedure in situations where licensee's programs or implementations of the programs were previously inspected and the licensee's performance indicates that additional review of previously inspected areas are not warranted or if regions elect to inspect licensee's implementation of the 10 CFR 50.59 program in phases (i.e., perform inspections of the licensee's procedures which will be used to implement the 10 CFR 50.59 program followed by another inspection of the licensee's implementation of their program).

END

Attachment:

Revision History for IP 37060 Attachment 1 Revision History For IP 37060 Issue Date: XX/XX/XX 1-1 37060

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	XX/XX/XX CN 10-XXX	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.”	Yes	N/A	ML103081058