# Program on Technology Innovation: 10CFR50.69 Implementation Guidance for Treatment of Structures, Systems, and Components

*Technical Report*

# Program on Technology Innovation: 10CFR50.69 Implementation Guidance for Treatment of Structures, Systems, and Components

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# REPORT SUMMARY

The subject of equipment categorization and treatment is very broad based and has been interpreted quite differently among engineering and quality assurance organizations at U.S. nuclear power plants over their operating lives. Most licensees have undergone several iterations of equipment categorization and classification, beginning with the requirements of 10CFR50 Appendix B, later through the implementation of numerous other regulations and industry guidance, and most recently to implement the guidance of INPO AP-913. This report presents each licensee with an opportunity to more clearly focus financial and human resources on components that matter most to the safe and reliable operation of their power plants.

## Background

On November 22, 2004, an important new rule, known as 10CFR50.69, was approved that presents nuclear management with an opportunity to further enhance equipment reliability and plant safety by focusing on those critical systems and components with the highest safety significance. The rule broadly adjusts the scope of safety-related components that are subject to the existing NRC regulations. The implementation of this new rule is strictly voluntary on the part of each licensee.

## Objectives

- To provide an overview of the 10CFR50.69 rule and how it should be implemented by interested licensees
- To provide guidance and reference to related industry documents to assist in the categorization of equipment in support of the new rule
- To provide clarification on achieving reasonable confidence once treatment is established for selected systems and components
- To provide guidance for establishing appropriate treatment for selected systems and components
- To provide examples of how implementation of the rule can affect treatment
- To provide a strategic transition strategy for licensees considering adoption of the new rule

## Approach

In cooperation with various EPRI targets (e.g. Risk and Safety Management, Plant Support Engineering, Non-Destructive Evaluation Center), a steering committee of utility engineers was formed. Utility membership was made up of representatives having experience with risk-informed categorization and treatment who could contribute lessons learned and benefits of the rule's intent. Other members were selected based upon their involvement with the two

companion documents addressing treatment guidance for low safety-significant equipment installed in seismic and harsh environmental applications. The steering committee had a dual purpose of not only developing the detailed guidance contained in this report, but also, at a more strategic level, coordinating with key industry organizations during the finalization of the rule and related supporting guidance documents.

The intent of this guide is to provide a consistent methodology for licensees who voluntarily adopt 10CFR50.69 to implement the treatment adjustments resulting from the completed categorizations. This report will be amended in the future with additional examples of how the rule can further impact treatment of selected components as additional implementation experience is gained throughout the industry.

## Results

This guide provides background behind the development of the 10CFR50.69 rule, including regulatory history, a brief introduction to the regulation, rule initiation, the exemption obtained by South Texas Project (STP), and pilot plants that have commenced categorization activities. Though not its main focus, the report provides a brief overview of categorization processes, including NEI 00-04 and the methodology used for categorization of structures, systems, and components (SSCs). Guidance is provided for interpreting the terms “reasonable confidence” and “reasonable assurance,” and the report provides a fundamental overview of design bases and licensing basis, and a generic process for developing and implementing treatment.

Structured similar to the rule, the report provides guidance for establishing alternative treatment with respect to inspections and tests, corrective action, feedback and change control, and administrative guidance in the areas of program documentation, change control, records retention, and reporting. An overall transition strategy for adopting 10CFR50.69, including a prioritized list of systems to which 10CFR50.69 can most cost-beneficially be applied, is also included.

## EPRI Perspective

This guide provides licensees with current guidance regarding the implementation of 10CFR50.69. It presents each licensee with an opportunity to more clearly focus financial and human resources on components that matter most to the safe and reliable operation of their power plants. The guidance in this report allows plant management to categorize equipment and adjust treatment so as to optimize maintenance activities, inventory, testing and inspection activities, and equipment availability while simultaneously increasing the safety of their plants through the application of risk-informed technology and probabilistic risk assessment.

## Keywords

10CFR50.69
Categorization
Reasonable confidence
Risk
Safety
Treatment

# EXECUTIVE SUMMARY

## Perceived Benefits

Adoption of 10CFR50.69 and subsequent risk-informed categorization will enable each licensee to more clearly identify and focus engineering, maintenance, and operations resources on those critical components with high safety significance. This ability to bring attention to those components (and subsequently spare/replacement items) will enable licensees to make better informed decisions, resulting in enhanced equipment reliability while simultaneously improving the safety of the their plant. Pilot implementation of 10CFR50.69 suggests that the rule can be a powerful way to enhance equipment reliability and plant safety, while enabling plant managers to apply resources to those components that truly matter most from both safety and generation perspectives.

## The Rule and Categorization

10CFR50.69, also known as Option 2, is a voluntary rule that improves plant safety, enhances equipment reliability, and focuses on a new risk-informed categorization process that results in four categories of structures, systems, and components (SSCs). In some cases, the resulting categorization removes low safety significant, safety-related components from the scope of the special treatment requirements imposed by NRC regulations. To distinguish what components are either safety significant or low safety significant, licensees must categorize components using an NRC-approved process. The Nuclear Energy Institute (NEI) has provided a comprehensive categorization guideline (NEI-00-04) that factors in probabilistic risk assessment (PRA) model insights as well as deterministic insights from an integrated decision-making panel (IDP). By blending the PRA and deterministic insights, an appropriately balanced, technically sound categorization result is produced. Any licensee using their existing PRA model can exercise this categorization process. The categorization output that results is used to determine the appropriate treatment that should be applied to components determined to be either safety significant or low safety significant.

Safety-related components determined to be safety significant are known as risk-informed safety classification- (RISC-) 1 and are treated no differently than they are today. Non-safety-related components determined to be safety significant are known as RISC-2 and are assessed to determine if any additional treatment is necessary.

Once categorized, safety-related components determined to be low safety significant are no longer required to be subject to the special treatment requirements imposed by the following NRC regulations:

- Quality Assurance requirements as defined in Appendix B
- 10CFR Part 21 reporting requirements
- Testing, documentation, and margin requirements for environmental qualification (EQ) purposes (10CFR50.49)
- Applicable portions of ASME and IEEE codes and standards (10CFR50.55a(f), (g), and (h))
- Maintenance Rule (10CFR50.65)
- Reporting requirement (10CFR50.72 and 50.73)
- Portions of Appendix J testing
- Seismic qualification with respect to the extent of testing and types of analyses (sections of Appendix A to 10CFR Part 100)

While the safety-related, low safety significant components (also known as RISC-3 components) are removed from the scope of certain regulatory requirements, 10CFR50.69 continues to impose some high-level treatment controls onto these components. These high-level treatment controls are intended to mirror existing balance of plant controls and provide “reasonable confidence” that these components will continue to satisfy their design functional requirements when required to do so.

Non-safety-related components determined to be low safety significant are known as RISC-4 and are treated no more rigorously than they are today.

It is important to note that 10CFR50.69 does not alter the design requirements and safety classification of any categorized component.

### The Rule and Treatment

10CFR50.69 imposes high-level treatment controls for categorized components, but largely relies on individual licensees to develop the details of how these controls are to be implemented. In order for licensees to consistently apply a “reasonable confidence” treatment approach, industry undertook the development of this report and complementary treatment guidelines to aid in this endeavor. This broad treatment guideline will continue to evolve as new industry insights and feedback are provided.

# ACKNOWLEDGMENTS


EPRI wishes to acknowledge the members of the Steering Committee for their valued contributions in the development of this report:

| | |
|---|---|
| John Winebrenner | Dominion Energy, Chairman |
| Glen Schinzel | South Texas Project, Vice Chairman |
| John Richards | Duke Energy |
| Maurice Dingler | Wolf Creek Nuclear Operating Company |
| Gary Toman | EPRI |
| Robert Kassawara | EPRI |

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# 1
# INTRODUCTION

## 1.1 Purpose

The purpose of this report is to provide licensees with guidance on how to implement 10CFR50.69. The implementation guidance focuses on the treatment of structures, systems, and components (SSCs) that have already been categorized from a risk-informed perspective by the licensee. Guidance concerning the categorization process is considered outside the scope of this report.

## 1.2 Report Scope

### *1.2.1 Relationship to the Rule and Other Guidance Documents*

Figure 1-1 illustrates how this report relates to the rule and other guidance documents. Requirements for implementing a risk-informed categorization and treatment of SSCs are described in 10CFR50.69, the adoption of which is optional for each licensee. The rule provides requirements for both phases of implementation—the categorization and the resulting treatment.

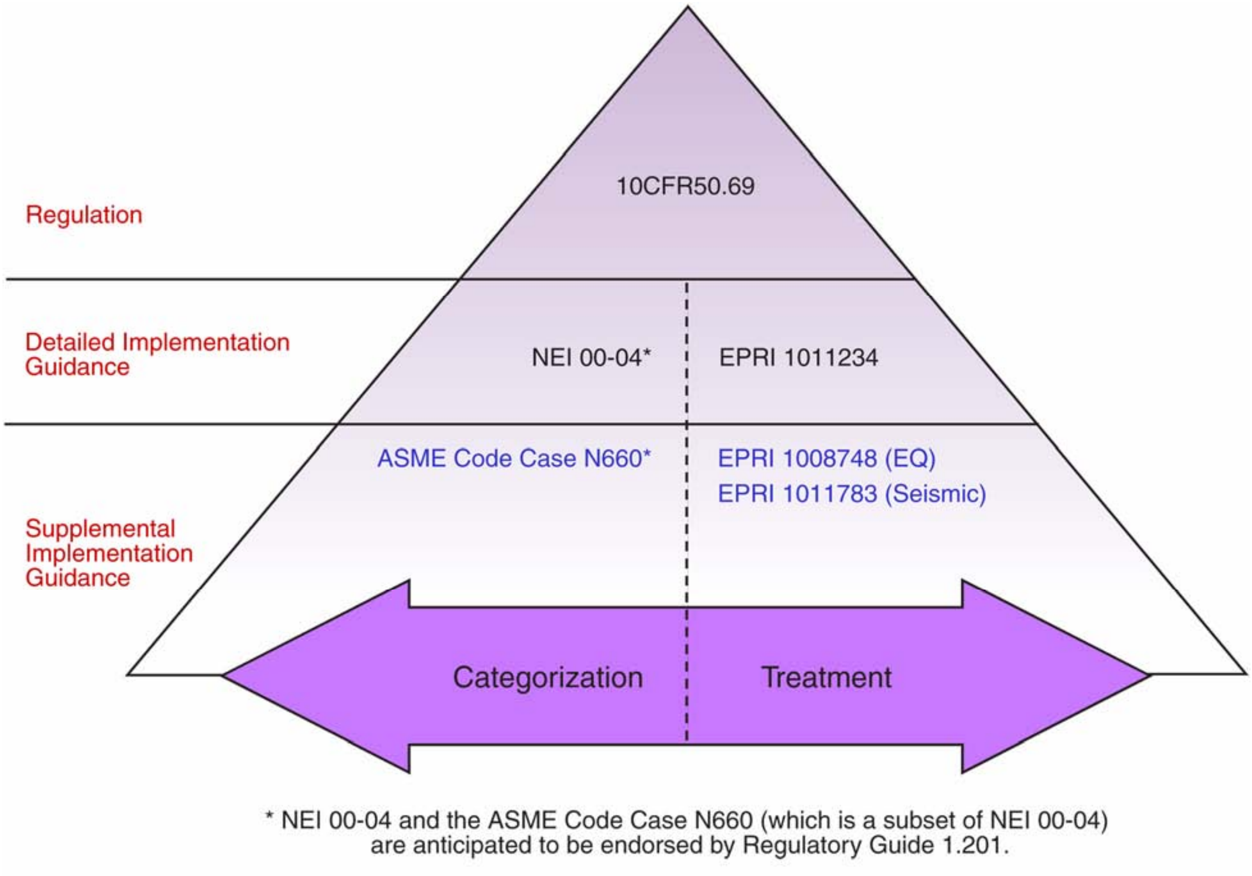


**Figure 1-1**
**Relationship with the Rule and Other Guidance Documents**

As shown in the figure, the licensee should use NEI 00-04, which is described briefly in Section 3.2 of this report, for guidance on implementing the risk-informed categorization process.

This report (EPRI 1011234) focuses on the treatment aspects of the rule and provides the licensee with guidance for interpreting the rule's treatment requirements and applying them to SSCs that have already been categorized.

Figure 1-1 also illustrates supplemental guidance documents that can assist the licensee with overall implementation of 10CFR50.69. The two EPRI reports noted (EPRI 1011783, *RISC-3 Seismic Assessment Guidelines*, and EPRI 1008748, *Guidance for Accident Function Assessment for RISC-3 Applications*) are companion documents to this report and provide specific detailed treatment guidance for seismically qualified and environmentally qualified components that have undergone the risk-informed categorization process.

The licensee should consider the guidance in ANS 3.2 for establishing high-level quality assurance requirements for SSCs categorized and treated under 10CFR50.69.

### *1.2.2 Basic Premises of This Report*

To put the guidance of this report in the proper context, the following basic premises should be considered:

- Categorization is assumed to have been conducted properly and in accordance with NEI 00-04. As such, this “treatment guide” contains only a brief overview of NEI 00-04 and points the reader to this and other related categorization documents for further details.
- For the purposes of this report, it is assumed that categorization of components with multiple functions (some safety significant and some low safety significant) are appropriately blended and considered to achieve a conservative RISC category result.
- While it is anticipated that categorization of SSCs may change over time, it is expected that recategorization will be infrequent (that is, will impact only a small number of SSCs). It is assumed that licensee feedback processes exist, consistent with NEI 00-04 requirements, to address insights that may result in changes in categorization. As such, this “treatment guide” provides only guidance and processes for addressing SSCs after they are initially categorized and issues associated with potential changes to treatment requirements as a result of SSCs changing from one category to another, for example, from RISC-3 to RISC-1.
- Adoption and implementation of 10CFR50.69 is completely optional for each licensee.
- This report presents guidance and minimal requirements (thresholds) to achieve reasonable confidence that RISC-2 and RISC-3 SSCs will perform their 10CFR50.69-credited functions. Licensees have the option to do more, at their discretion.

### *1.2.3 Report Structure and Organization*

#### 1.2.3.1 Current Structure and Organization

Section 2 of the report provides the reader some background behind the development of the 10CFR50.69 rule, including regulatory history, a brief introduction to the regulation, rule initiation, the exemption obtained proof-of-concept demonstration by South Texas Project (STP), and pilot plants that have begun categorization activities. Section 3 of the report provides a brief overview of categorization processes, including NEI 00-04 and the methodology used for categorization of SSCs. Section 4 discusses the terms “reasonable confidence” and “reasonable assurance,” and how a licensee adopting 10CFR50.69 can, in a generic sense, obtain an appropriate level of reasonable confidence that SSCs will satisfy design functional requirements.

Section 5 provides a fundamental overview of design bases and licensing basis, and Section 6 presents a generic process for developing and implementing treatment for each of the RISC categories, with special emphasis on RISC-3 SSCs.

During the operating life of a RISC-3 SSC, it is necessary to have a sufficient level of confidence that the SSC will continue to be able to perform and satisfy its design basis functions; hence, Section 7 includes guidance for inspections and tests. Attachment A of the report is an illustrative example of how treatment can be developed using the generic process presented in

Section 6, and applied to a common testing program at nuclear power plants (local leak-rate testing).

Once test and inspection data are collected, they should be fed back into the categorization and treatment processes, and when important deficiencies are found, they should be corrected. Consequently, corrective action guidance is also provided in these areas and is also found in Section 7. Section 8 provides guidance on feedback and change control, and Section 9 provides administrative guidance in the areas of program documentation, change control, records retention, and reporting.

Section 10 of this report describes an overall transition strategy for adopting 10CFR50.69, including a prioritized list of systems to which 10CFR50.69 can most cost-beneficially be applied. References are provided in Section 11.

Examples of assessing the adequacy of treatment for RISC-2 items is provided in Appendix B, and guidance regarding the impact the rule has on current procurement processes is discussed in Appendix C of the report, along with some illustrative examples of procuring RISC-3 items. Finally Appendix D discusses the applicability of guidance contained in Generic Letter 91-18 regarding operability.

#### 1.2.3.2 Anticipated Structure and Organization

The development of this report coincided with the finalization of not only the rule but also NEI and ASME guidance. Its publication was scheduled so as to provide timely guidance to licensees considering the adoption and implementation of the rule. The long-term intent of this report is to add other appendices that will provide additional examples of the rule's implementation and establish treatment that is deemed appropriate and sufficient to provide reasonable confidence in the performance of SSC design functions.

## 1.3 Perceived Benefits of Adopting 10CFR50.69

### *1.3.1 Background and Relationship to INPO AP-913*

INPO AP-913, "Equipment Reliability Process Description" requires licensees to identify critical components and subsequently categorize equipment as critical, non-critical, or run-to-failure. Equipment categorization per INPO AP-913 considers many criteria, one of which is each item's deterministic safety classification. For most licensees, categorization of components per INPO AP-913 began in or around 2003 and, as such, would precede any risk-informed categorization for licensees considering adoption of 10CFR50.69.

In most cases, categorization under INPO AP-913 conservatively assumed that all deterministically classified "safety-related" components were also "critical components." The components that were subsequently categorized as "critical" encompassed more than just safety-related items. The critical component category also includes those components important to generation and plant reliability.

### 1.3.2 Key Benefit of 10CFR50.69

Adoption of 10CFR50.69 and subsequent risk-informed categorization enables each licensee to identify and more clearly focus engineering, maintenance and operations resources on those critical components with high safety-significance. This ability to bring attention to those components (and subsequently spare/replacement items) will enable licensees to make better informed decisions resulting in enhanced equipment reliability while simultaneously improving the safety of the their plant. Pilot implementation of 10CFR50.69 suggests that the rule can be a powerful way to enhance equipment reliability and plant safety, while enabling plant managers to more appropriately apply resources to those components that truly matter from both safety and plant generation perspectives.

As noted in the previous section, adoption and implementation of the rule is completely optional for the licensee.

Figure 1-2 illustrates the key points discussed in the preceding sections and shows how the rule can first identify (through proper categorization) safety-significant items by evaluating the population of components that would have been identified as “critical” upon the completion of INPO AP-913 implementation. Once the new population of safety-significant items is identified, then resources can be applied and focused as a way to achieve higher equipment reliability, plant availability, and overall plant safety.

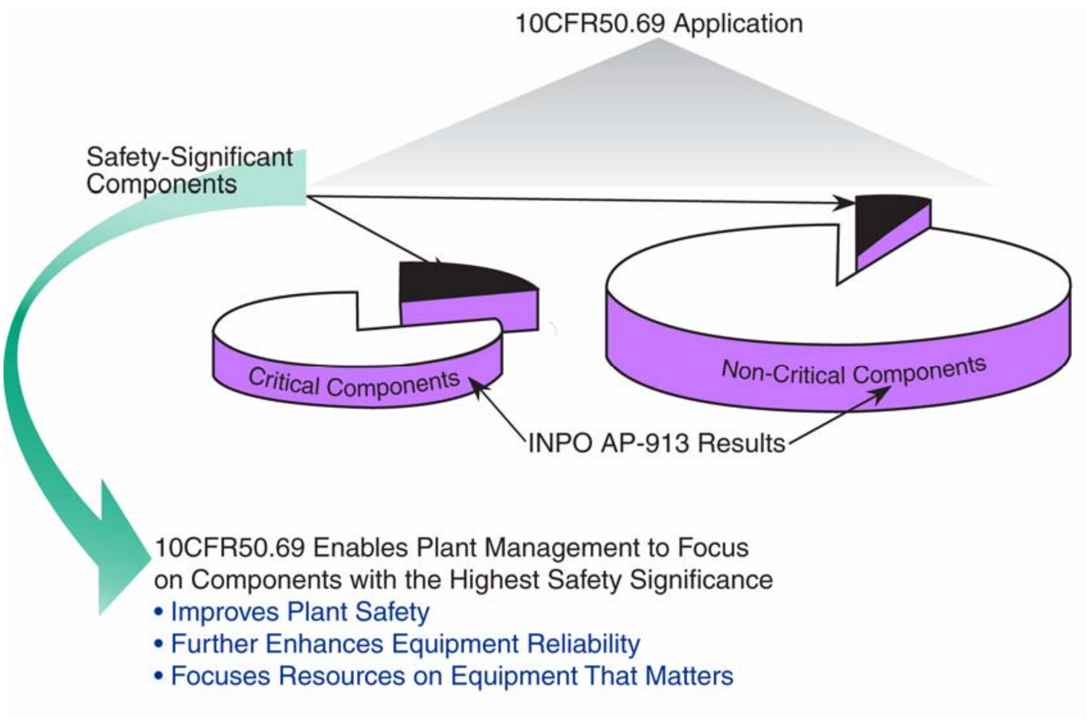


**Figure 1-2**
**Potential Benefits of Adopting 10CFR50.69**

## 1.4 Definitions of Key Terms

Reasonable assurance – A justifiable level of confidence based on objective and measurable facts, actions, or observations, which infer adequacy (Reference EPRI TR-102260).

Reasonable confidence – A level of confidence based on facts, actions, knowledge, experience, and/or observations, which are deemed to be adequate (Reference EPRI 1011234).

Risk-informed safety categorization – The categorization process that places each SSC in one of the following four risk informed safety categories (Reference 10CFR50.69):

- RISC-1 – SSCs that are safety related and perform safety-significant functions
- RISC-2 – SSCs that are non-safety related and perform safety-significant functions
- RISC-3 – SSCs that are safety related and perform low safety-significant functions
- RISC-4 – SSCs that are non-safety related and perform low safety-significant functions.

Safety-significant function – A function whose degradation or loss could result in a significant adverse affect on defense-in-depth, safety margin, or risk. (Reference 10CFR50.69)

Special treatment requirements – Prescriptive requirements as to how licensees are to treat SSCs, especially those that are defined as "safety-related" (Reference 10CFR50.69)

Treatment – Activities, processes, and/or controls that are performed or used 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 assessment and feedback of results to adjust activities as needed to meet desired outcomes.

## 1.5 Acronyms and Abbreviations

ANI – Authorized Nuclear Inspector

ANPR – Advance Notice of Proposed Rulemaking

ASME – American Society of Mechanical Engineers

ATWS – anticipated transient without scram

CDF – core damage frequency

CFR – Code of Federal Regulations

CGI – commercial grade item

CIV – containment isolation valve

DCP – design change package

EOP – emergency operating procedures

EQ – environmental qualification

FMEA – failure modes and effects analysis

FSAR – final safety analysis report

GDC – general design criteria

GIP – generic implementation procedure

HCPI – high-pressure coolant injection

HEP – human error probability

HSS – high safety significant

IDP – integrated decision-making panel

ILRT – integrated leak rate test(ing)

IPE – individual plant evaluation

IRC – inside-reactor-containment

LCO – limiting condition for operation

LERF – large early release frequency

LLRT – local leak rate test(ing)

LOCA – loss of coolant accident

LSS – low safety significant

NEI – Nuclear Energy Institute

OQAP – Operational Quality Assurance Program

PM – preventive maintenance

PMT – post-maintenance test

PORV – pressure-operated relief valve

PRA – probabilistic risk assessment

PWR – pressurized water reactor

QA – quality assurance

RG – Regulatory Guide

RISC – risk-informed safety categorization (Note: 10CFR50.69 also uses this acronym to represent risk-informed safety class.)

RPV – reactor pressure vessel

RTS – repetitive task sheet

SBO – station blackout

SECY – Commission Papers

SOC – Statement of Consideration

SQUG – Seismic Qualification Utility Group

SQURTS – Seismic Qualification Reporting and Testing Standardization

SRM – staff requirements memorandum

SSC – structure, system, and component

STPNOC – South Texas Project Nuclear Operating Company

TERI – technical evaluation of replacement items

TR – technical report

UFSAR – updated final safety analysis report

# 2
# BACKGROUND

## 2.1 Regulatory History

### *2.1.1 Introduction to the Regulation*

10CFR50.69 represents an alternative set of requirements whereby a licensee or applicant may voluntarily undertake categorization of its SSCs consistent with the requirements in § 50.69(c), remove the special treatment requirements listed in § 50.69(b) for SSCs that are determined to be of low individual safety significance, and implement alternative treatment requirements in § 50.69(d). The regulatory requirements not removed by § 50.69(b) continue to apply as well as the requirements specified in § 50.69. The rule 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, a risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, which uses both risk insights and traditional engineering insights. The safety functions include both the design basis functions (derived from the "safety-related" definition, which includes external events), as well as functions credited for severe accidents (including external events).

Treatment for the SSCs is required to be applied as necessary to maintain functionality and reliability, and is a function of the category into which the SSC is placed. Finally, assessment activities are conducted to make adjustments to the categorization and treatment processes as needed so that SSCs continue to meet applicable requirements.

The rule is comprised of the following major sections:

a. Definitions

b. Applicability and scope of risk-informed treatment of SSCs and submittal/approval process

c. SSC categorization process

d. Alternative treatment requirements

e. Feedback and process adjustment

f. Program documentation, change control, and records

g. Reporting

The adoption of the rule is completely voluntary. When the rule is adopted and the licensee commits to categorization and treatment in accordance with the rule, the rule requires that the entire population of components within a system is considered and categorized, whether currently safety related or not. However, the licensee does have latitude in selecting those systems to which the rule will apply. Through categorization, some components that are currently safety related might be found to support only low safety-significant functions, and some current non-safety-related components might be found to be safety significant.

### *2.1.2 Rule Initiation*

In addition to Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," the NRC also issued other regulatory guides on risk-informed approaches for specific types of applications. These included:

- RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing"
- RG 1.176, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance"
- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"
- RG 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Inservice Inspection of Piping"

In this respect, the Commission has been successful in developing and implementing a regulatory means for considering risk insights into the current regulatory framework. One such risk-informed application, the South Texas Project (STP) submittal on graded quality assurance is particularly noteworthy.

## 2.2 South Texas Project (STP) Proof of Concept Demonstration

In March 1996, STP Nuclear Operating Company (STPNOC) requested that the NRC approve a revised Operations Quality Assurance Program (OQAP) that incorporated the methodology for grading quality assurance (QA) activities based on PRA insights. The STP graded QA proposal was an extension of the existing regulatory framework. Specifically, the STP approach continued to use the traditional safety-related categorization, but it allowed for gradation of safety significance within the "safety-related" categorization (consistent with 10CFR50 Appendix B) through the use of a risk-informed process. Following extensive discussions with the licensee and substantial review, the NRC staff approved the proposed revision to the OQAP on November 6, 1997.

Subsequent to the NRC's approval, STPNOC identified implementation difficulties associated with the graded QA program. Despite the reduced QA requirement applied for a large number of SSCs that the licensee judged to be of low safety significance, other regulatory requirements (such as environmental qualification, the Boiler and Pressure Vessel Code (BPV) of the American Society of Mechanical Engineers (ASME), or seismic requirements) continued to impose substantial burdens. As a result, the replacement of a low-safety-significant component needed to satisfy other special requirements during the procurement process. These requirements prevented STPNOC from realizing the full potential reduction in unnecessary regulatory burden for SSCs judged to have little or no safety importance. In an effort to achieve the full benefit of the graded QA program (and, in fact, to go beyond the staff's previous approval of graded QA), STPNOC submitted a request, dated July 13, 1999, asking for an exemption from the scope of numerous special treatment regulations (including 10CFR50 Appendix B) for SSCs categorized as low safety significant or as non-risk significant. STPNOC's exemption was ultimately approved by the staff in August 2001.

The experience with graded QA was a principal factor in the NRC's determination that rule changes would be necessary to fully implement the benefits of risk-informed technology. The Commission also believed that the development of PRA technology and decision-making tools for using risk information, together with deterministic information, supported rulemaking activities that would allow the NRC to refocus certain regulatory requirements using this type of information.

Under Option 2 of SECY–98–300, "Options for Risk-Informed Revisions to 10 CFR Part 50—'Domestic Licensing of Production and Utilization Facilities,'" dated December 23, 1998, the NRC staff recommended that risk-informed approaches to the application of special treatment requirements be developed as one application of risk-informed regulatory changes. Option 2 (also referred to as RIP50 Option 2) addresses the implementation of changes to the scope of SSCs needing special treatment while maintaining the existing regulatory rule language. Under Option 2, the licensee must still provide assurance that the SSCs will perform their design functions. Changes to the requirements pertaining to the design basis functional requirements of the plant or the design basis accidents are not included in Option 2. These technical risk-informed changes are addressed under Option 3 of SECY–98–300. The Commission approved proceeding with Option 2 in a staff requirements memorandum (SRM) dated June 8, 1999.

The stated purpose of the Option 2 rulemaking was to develop an alternative regulatory framework that enables licensees, using a risk-informed process for categorizing SSCs according to their safety significance (that is, a decision that considers both traditional deterministic insights and risk insights), to reduce unnecessary regulatory burden for SSCs of low safety significance by removing these SSCs from the scope of special treatment requirements. As a result, both the NRC and industry should be able to better focus their resources on regulatory issues of greater safety significance.

The Commission directed the NRC staff to evaluate strategies to make the scope of the nuclear power reactor regulations that impose special treatment risk-informed. SECY–99–256, "Rulemaking Plan for Risk-Informing Special Treatment Requirements," dated October 29, 1999, was sent to the Commission to obtain approval for a rulemaking plan and issuance of an

Advance Notice of Proposed Rulemaking (ANPR). By SRM dated January 31, 2000, the Commission approved publication of the ANPR and approved the rulemaking plan. The ANPR was published in the Federal Register on March 3, 2000 (65 FR 11488), for a 75-day comment period, which ended on May 17, 2000. In the rulemaking plan, the NRC proposed to create a new section within part 50, now identified as § 50.69, to contain these alternative requirements.

The Commission received more than 200 comments in response to the ANPR. The NRC staff sent the Commission SECY–00–0194, “Risk-Informing Special Treatment Requirements,” dated September 7, 2000, which provided the staff’s preliminary views on the ANPR comments and additional thoughts on the preliminary regulatory framework for implementing a rule to revise the scope of special treatment requirements for SSCs. The comments from the ANPR are further discussed in Section IV.1.0 of SECY–02–0176, “Proposed Rulemaking to Add New Section 10 CFR50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems, and Components,’” dated September 30, 2002 (ADAMS accession number ML022630007).

The concept developed for this rule, discussed at length in the ANPR, applies treatment requirements based upon the safety significance of SSCs, determined through consideration of both risk insights and deterministic information. Thus, the risk-informed approach discussed in this rule for establishing an alternative scope of SSCs subject to special treatment requirements uses both risk and traditional deterministic methods in a blended “risk-informed” approach.

The NRC staff prepared a proposed rule package and provided it to the Commission in SECY–02–0176. The Commission approved issuance of proposed 10CFR50.69 for public comment in an SRM dated March 28, 2003. The proposed 10CFR50.69 rule was published for public comment in the *Federal Register* on May 16, 2003 (68 FR 26511). The Commission received 26 sets of comments in response to the proposed rule. The comments are discussed in Section II of the Statements of Consideration.

The NRC staff provided the draft final rule to the Commission in SECY–04–0109 dated June 30, 2004. The Commission subsequently approved the final rule subject to the changes denoted during the session and documented in the SRM dated October 7, 2004 (ADAMS accession number ML042810516). The final rule was issued November 22, 2004.

## 2.3 Industry Pilot Plant Activities

At the time this report was published, two nuclear power plants had performed preliminary risk-informed categorizations of selected plant systems:

- Surry Nuclear Power Station – Dominion
- Wolf Creek Power Station – Wolf Creek Nuclear Operating Company (WCNOC)

These initial categorization efforts were performed as pilot activities to demonstrate the risk-informed categorization process and the resulting cost benefits of adopting 10CFR50.69.

# 3
# CATEGORIZATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

## 3.1 Requirements of 10CFR50.69

### *3.1.1 General Requirements*

10CFR50.69 defines four RISC categories into which SSCs are categorized. Four categories were chosen because it is the simplest approach for transitioning between the previous SSC classification scheme and the new scheme used in § 50.69. Figure 3-1 provides a conceptual understanding of the new RISC categories and depicts how NEI 00-04 can be used to categorize SSCs from a risk-informed perspective. The top portion of the figure illustrates how, in the traditional deterministic approach, SSCs were generally categorized as either safety related (as defined in § 50.2) or non-safety related. This division is shown by the vertical dotted line in the oval. Risk insights, including consideration of severe accidents, can be used to identify SSCs as being safety significant or low safety significant (shown by the horizontal line in the lower portion of the figure). Hence, the application of a risk-informed categorization results in SSCs being grouped into one of four categories as represented by the four boxes in the lower portion of Figure 3-1.

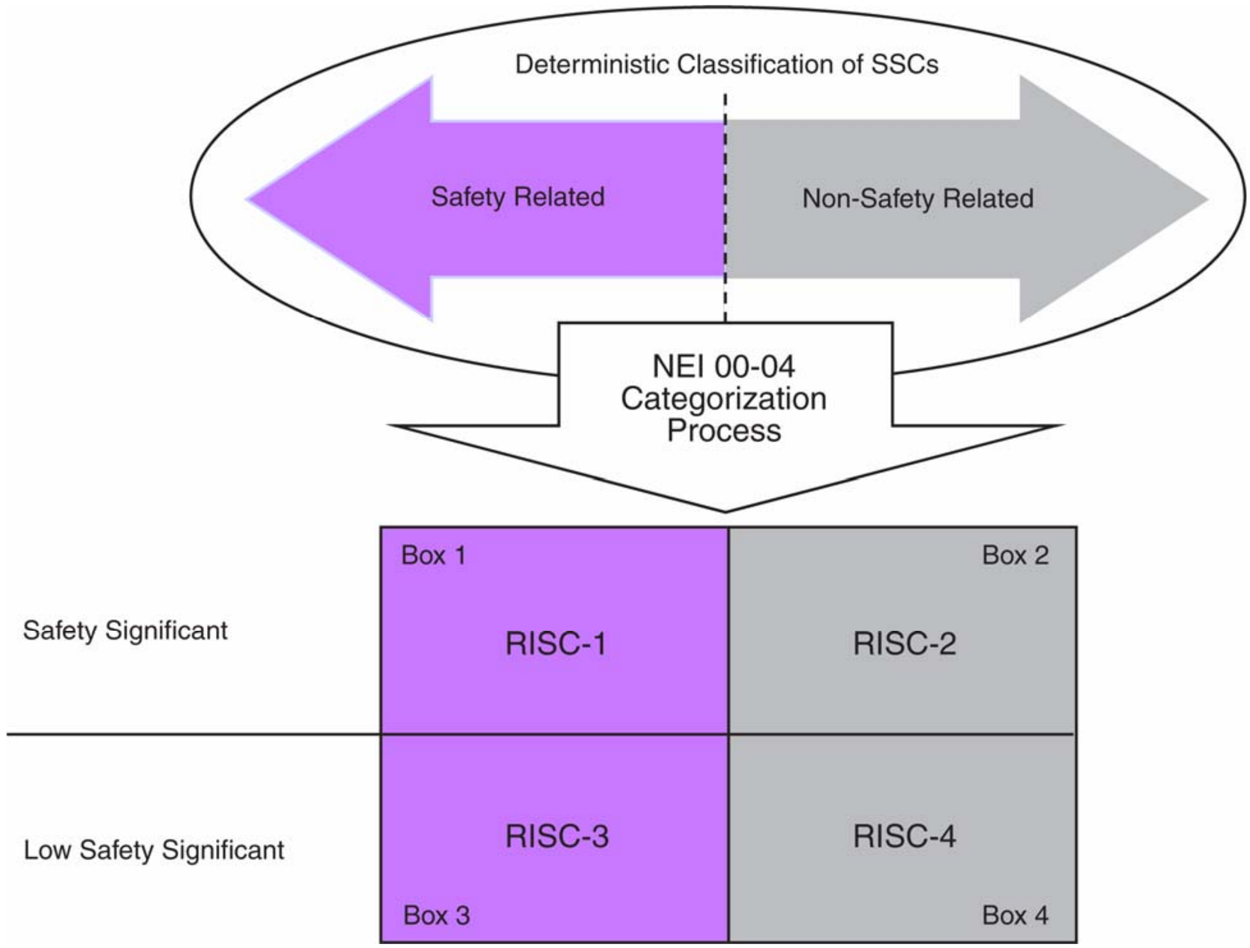


**Figure 3-1**
**Risk-Informed Safety Categorizations (RISC)**
**Courtesy of 10CFR50.69 and NEI-00-04**

Box 1 of Figure 3-1 depicts safety-related SSCs that a risk-informed categorization process determines are significant contributors to plant safety. These SSCs are termed RISC-1 SSCs. RISC-2 SSCs, depicted by Box 2 in Figure 1, are non-safety-related SSCs that the risk-informed categorization determines to be significant contributors to plant safety. RISC-3 SSCs are those SSCs that are safety related and that a risk-informed categorization process determines are not significant contributors to plant safety. These SSCs are depicted by Box 3 in Figure 3-1. Finally, there are SSCs that are non-safety related and that a risk-informed categorization process determines are not significant contributors to plant safety. These are termed RISC-4 SSCs and are depicted by Box 4 in Figure 3-1.

10CFR50.69 defines the terminology “safety-significant function” as a function whose loss or degradation could have a significant adverse effect on defense in depth, safety margins, or risk. This definition was chosen to be consistent with the concepts described in RG 1.174. The rule maintains more treatment requirements on SSCs that perform safety-significant functions than on SSCs that perform only low safety-significant functions to ensure that risk is low and that defense in depth and safety margins are maintained. As such, more treatment would be expected on RISC-1 SSCs than on RISC-3 SSCs, and similarly more treatment would be expected on RISC-2 SSCs than on RISC-4 SSCs. The rule also requires that the licensee or applicant provide reasonable confidence that the change in risk associated with implementation of § 50.69 will be small.

### *3.1.2 Methodology for Categorization*

The rule provides definitive guidance regarding the methodology for categorization, which should be considered in conjunction with NEI 00-04. The rule states the following:

> (1) SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines if an SSC performs one or more safety significant functions and identifies those functions. The process must:
>
> (i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.
>
> (ii) Determine SSC functional importance 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. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. 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.
>
> (iii) Maintain defense-in-depth.
>
> (iv) 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 permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.
>
> (v) Be performed for entire systems and structures, not for selected components within a system or structure.
>
> (2) The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant- knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

Thus, regarding categorization, the rule emphasizes the following two key points:

- SSCs must be categorized using a process that determines if an SSC performs one or more safety-significant functions and that must identify those functions.
- SSCs must be categorized by an integrated decision-making panel (IDP).

NEI 00-04 defines the categorization process that licensees should use to meet the requirements of the rule stated above.

## 3.2 Overview of Categorization Processes

### *3.2.1 Introduction*

Figure 3-2 illustrates the general scope of components that may be categorized under 10CFR50.69. The licensee should use NEI 00-04 for categorizing components with active functions and ASME Code Case N660 for pressure-boundary components with passive functions.

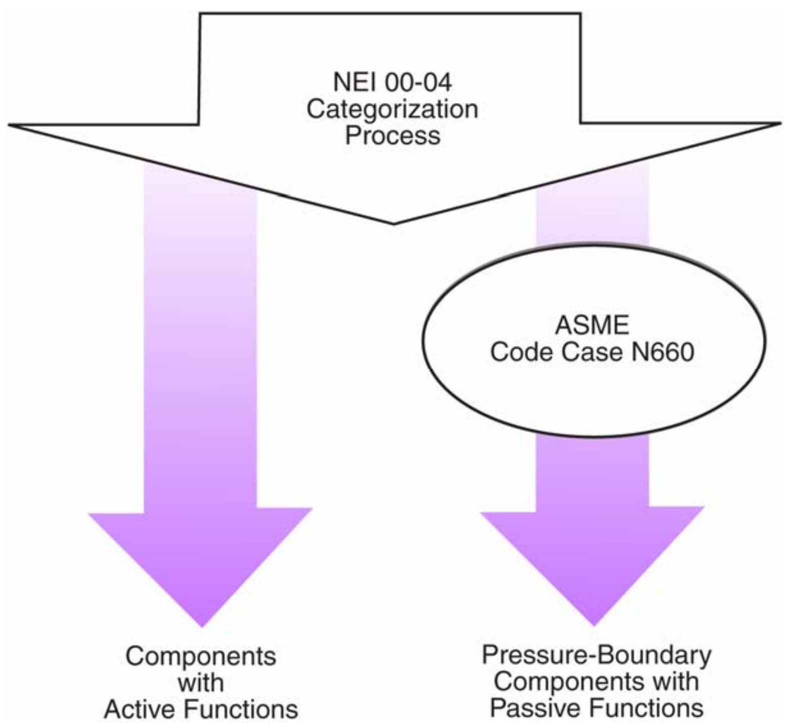


**Figure 3-2**
**Scope of Components Categorized Under 10CFR50.69**

Because the ASME Code Case is referenced in the NEI document, for simplicity in this report, only the NEI report is cited. The reader should recognize that, in fact, both documents are being referenced.

### 3.2.2 NEI 00-04

The categorization process described in NEI 00-04 uses a series of evaluations to determine the proper risk-informed safety classification for structures, systems, and components (SSCs). The overall process involves:

- A risk characterization of the safety significance of all SSCs in a plant system
- A defense-in-depth characterization to ensure that adequate redundancy and diversity for design bases events are maintained
- An integrated risk sensitivity study to ensure that any potential increases in risk are small
- Presentation of the results of these evaluations to an integrated decision-making panel (IDP) that determines the final categorization of the SSCs

The NEI 00-04 categorization process addresses a full scope of hazards, as well as plant shutdown safety. Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the safety significance of SSCs is assessed separately from each of five perspectives:

- Internal event risks
- Fire risks
- Seismic risks
- Other external risks (for example, tornadoes, external floods, etc.)
- Shutdown risks

The categorization process results in the four categories noted in 10CFR50.69:

- RISC-1 - SSCs that are safety related and perform safety-significant functions
- RISC-2 - SSCs that are non-safety related and perform safety-significant functions
- RISC-3 - SSCs that are safety related and perform low safety-significant functions
- RISC-4 - SSCs that are non-safety related and perform low safety-significant functions.

For components that are determined to have RISC-3 applications, the special treatments such as seismic and environmental qualification no longer apply. In lieu of these special treatments, licensees must implement alternative treatments that provide “reasonable confidence” that the RISC-3 devices will perform their required safety functions. This report provides guidance for establishing reasonable confidence that RISC-3 components can perform their safety functions under applicable accident functions.

#### 3.2.2.1 NEI 00-04 Guiding Principles

The principles for categorizing SSCs have been assessed through pilot plant implementation and are:

- Use applicable risk assessment information.
- Use deterministic or qualitative information if no PRA information exists related to a particular hazard or operating mode.
- Use a blended approach considering both quantitative PRA information and qualitative information for the categorization process.
- Maintain the Regulatory Guide 1.174 principles of the risk-informed approach to regulations.
- Categorize a safety-related SSC as RISC-1 unless a basis can be developed for categorizing it as RISC-3.
- Document attributes that make an SSC safety-significant.

#### 3.2.2.2 NEI-00-04 Voluntary and Selective Implementation

U.S. nuclear generating plants have attained and maintained an outstanding safety performance record. The existing NRC regulations together with the NRC’s regulatory oversight and inspection processes clearly provide adequate protection of public health and safety. As a result, the decision to focus on equipment that matters most from both a safety and equipment reliability perspective is a voluntary licensee decision. Each licensee should make its determination to adopt the new rule based on the estimated safety and economic benefits.

From a safety perspective, the benefits are associated with a better licensee and NRC focus of attention and resources on matters that are safety significant. A risk-informed SSC categorization process should result in an increased awareness on that set of equipment and activities that could impact safety and, hence, an overall improvement in safety.

### *3.2.3 ASME Code Case N660*

As stated in ASME Code Case N-660, this Case provides classification criteria that may be used as a supplement to the group classification criteria of IWA-1320 to determine risk-informed safety classification for use in risk-informed repair/replacement activities. The Case uses the term “classification” in lieu of “categorization” as does 50.69. Also, the Case uses “high safety significant” in lieu of “safety significant.” The RISC process of this case may be applied to Class 1, 2, 3, or non-class pressure-retaining items or their associated supports, except core supports, in accordance with the risk-informed safety classification criteria established by the regulatory authority having jurisdiction at the plant site. Non-class items are items not classified in accordance with IWA-1320.

The RISC process is described in Appendix I of the Case. Pressure-retaining and component support items are classified high safety significant (HSS) or low safety significant (LSS). The licensee should note that these classifications/categorizations might not be directly related to other risk-informed applications.

The case also states that Class 1 items that are part of the reactor coolant pressure boundary except as provided in paragraphs (c)(2)(i) and (c)(2)(ii) of 10CFR50.55a must be classified high safety significant. For items that are connected to the reactor coolant pressure boundary, as defined in paragraphs (c)(2)(i) and (c)(2)(ii) of 10CFR50.55a, the RISC process as described in Appendix I of the Case should be applied.

## 3.3 Other Guidance Regarding Equipment Categorization

### *3.3.1 INPO AP-913*

As discussed in Section 1 of this report, INPO AP-913 establishes a high-level process for scoping and identifying of critical components. Implementation guidance has been provided in EPRI Report 1007935, *Critical Component Identification Process – Licensee Examples (Scoping and Identification of Critical Components in Support of INPO AP-913).* Adoption of 10CFR50.69 suggests that it can enhance the effectiveness of INPO AP-913 implementation by enabling plant management to focus on safety-significant critical components and further improving both equipment reliability and plant safety.

### *3.3.2 Other Categorization Processes*

Through the initiation of various nuclear regulatory guides, industry initiatives, and guidance implementation, licensees have identified and categorized certain components affected by the particular rule or guidance. In effect, the licensee was required to programmatically categorize equipment so the treatment and regulatory requirements could be applied consistently. These other program categorizations are not explicitly impacted when adopting 10CFR50.69; however, the scope of SSCs subject to these regulatory programs will likely be adjusted after completion of the 10CFR50.69 categorization process. As shown in Section 6 of this report, the licensee has the opportunity to change treatment for these program SSCs if they have been categorized RISC-3 because the programs that initially identified and categorized those RISC-3 SSCs are no longer required to be imposed (that is, Maintenance Rule, IST, ISI, AOV, MOV, etc.).

## 3.4 Quantitative Results of Categorization Efforts

The following quantitative results were provided by the three utilities that have implemented the risk-informed categorization process described in NEI 00-04: South Texas Project, Surry Power Station, and Wolf Creek Power Station. The categorization results are provided for illustrative purposes only and to demonstrate the degree to which each IDP categorized existing system components. These results should be considered representative of the type of results that can be achieved at other nuclear power plants.

Figures 3-3 through 3-14 illustrate how the categorization process described in 10CFR50.69 and implemented via NEI 00-04 was applied to various plant operating systems and how the process resulted in many components being categorized as low safety significant (RISC-3 or RISC-4).

The left side of each figure represents the original deterministic safety classification for the components within a given system. As such, the two bars on the left side of each figure represent the relative number of safety-related and non-safety-related components.

The right side of each figure represents the resulting risk-informed categorization that resulted after implementing NEI 00-04. In each case, the number of safety or non-safety components determined to be safety significant are shown near the bottom of each of the two bars. The number of safety or non-safety components found to be low safety significant is shown on the top portion of each bar in white.

The vertical axis on each side of the figures is scaled proportionately and depicts the total number of components categorized in each respective system.

Guidance regarding the appropriate treatment of items categorized into each of the four risk-informed categories is provided in Sections 4 through 8 of this report. Examples of treatment applied to various categories of items are also provided in the appendices of this report.

### *3.4.1 Summary of Categorization Results from STP*

Figure 3-3 illustrates the overall results achieved at South Texas Project. The figure shows that after risk-informed categorization, only 24% of the safety-related SSCs were categorized as safety significant, whereas 76% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 1% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

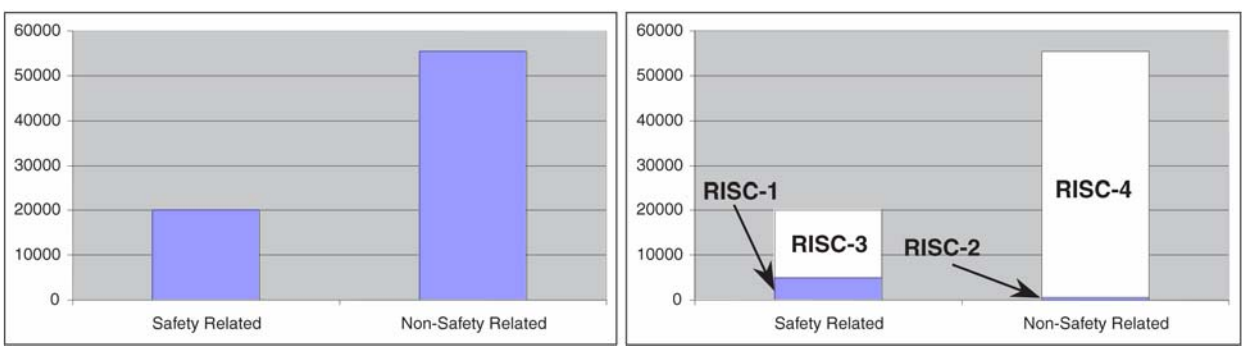


**Figure 3-3**
**Scope of Components Categorized Under 10CFR50.69**

### *3.4.2 Reactor Coolant System Categorization Results from STP*

Figure 3-4 illustrates the categorization results achieved at South Texas Project for SSCs in the reactor coolant system (RCS). The figure shows that after risk-informed categorization, only 66% of the safety-related SSCs remained safety significant, whereas 34% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 14% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

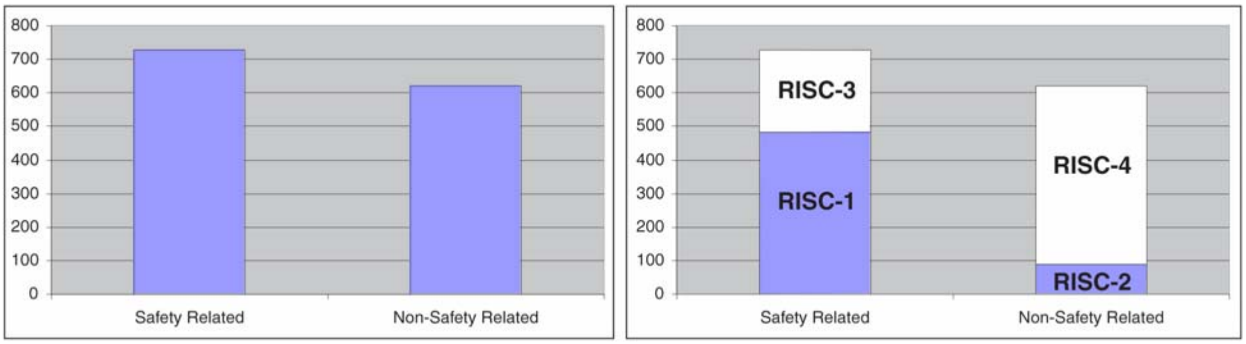


**Figure 3-4**
**Reactor Coolant System SSCs Categorized Under 10CFR50.69**

### *3.4.3 Auxiliary Feedwater System Categorization Results from STP*

Figure 3-5 illustrates the categorization results achieved at South Texas Project for SSCs in the auxiliary feedwater system. The figure shows that after risk-informed categorization, only 32% of the safety-related SSCs were categorized as safety significant, whereas 68% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 6% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

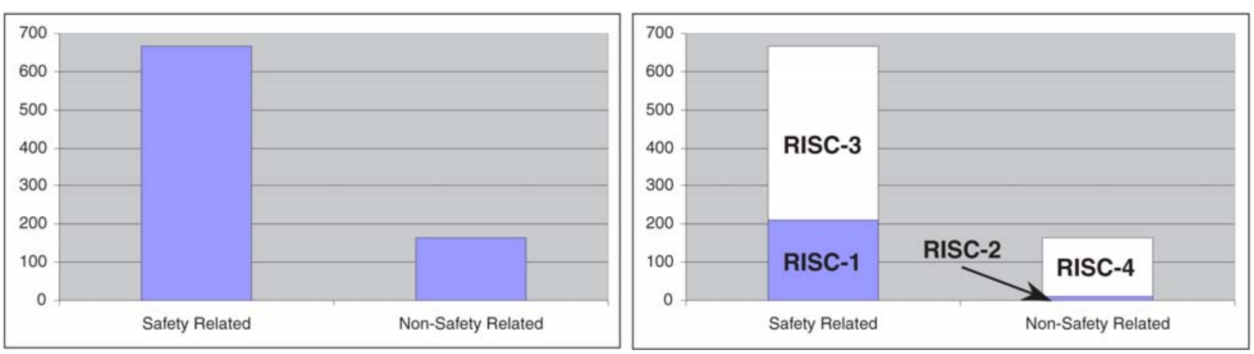


**Figure 3-5**
**Auxiliary Feedwater System SSCs Categorized Under 10CFR50.69**

### *3.4.4 Residual Heat Removal System Categorization Results from STP*

Figure 3-6 illustrates the categorization results achieved at South Texas Project for SSCs in the residual heat removal system. The figure shows that after risk-informed categorization, only 36% of the safety-related SSCs were categorized as safety significant, whereas 64% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 20% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

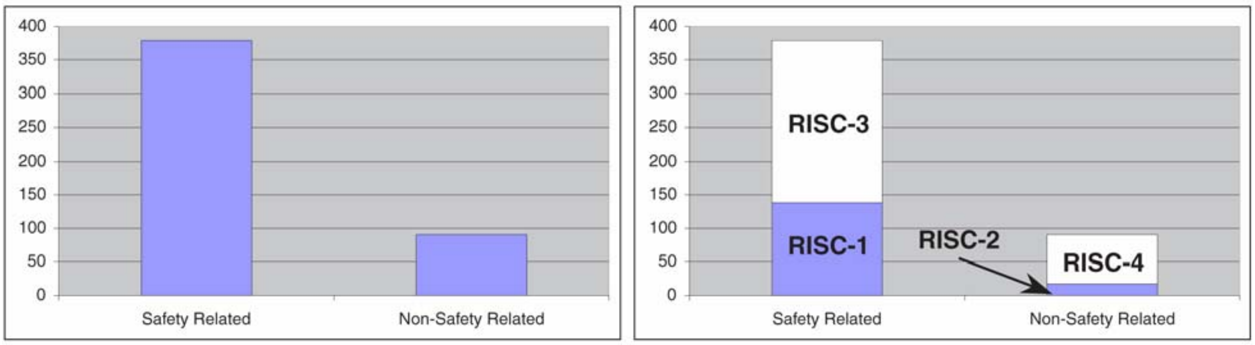


**Figure 3-6**
**Residual Heat Removal SSCs Categorized Under 10CFR50.69**

### *3.4.5 Control Building Ventilation System Preliminary Active SSC Categorization Results from Wolf Creek*

Figure 3-7 illustrates preliminary categorization results achieved at Wolf Creek for active SSCs in the control building ventilation system. The figure shows that after risk-informed categorization, only 36% of the safety-related SSCs were categorized as safety significant, whereas 64% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 20% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

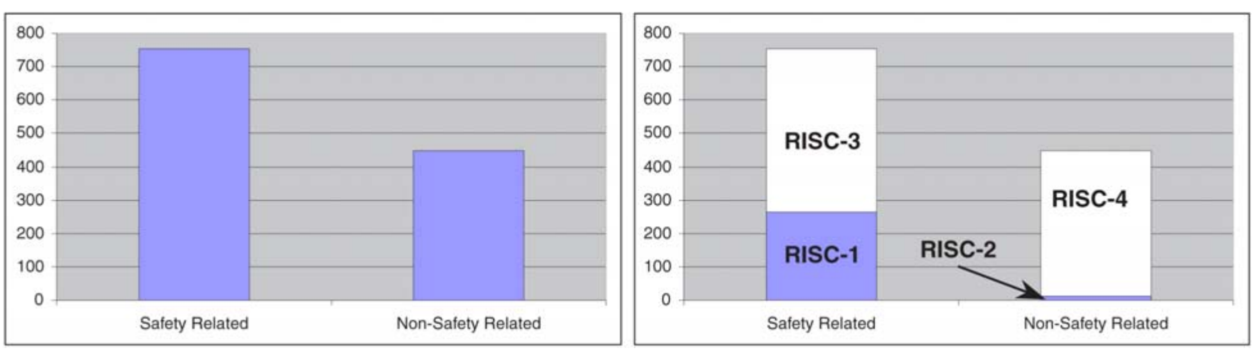


**Figure 3-7**
**Control Building Ventilation System Active SSCs Categorized Under 10CFR50.69**

### 3.4.6 Control Building Ventilation System Preliminary Passive SSC Categorization Results from Wolf Creek

Figure 3-8 illustrates preliminary categorization results achieved at Wolf Creek for passive SSCs in the control building ventilation system. The figure shows that after risk-informed categorization, none of the safety-related SSCs were categorized as safety significant, whereas 100% of those safety-related SSCs were categorized as low safety significant. The figure also shows that none of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

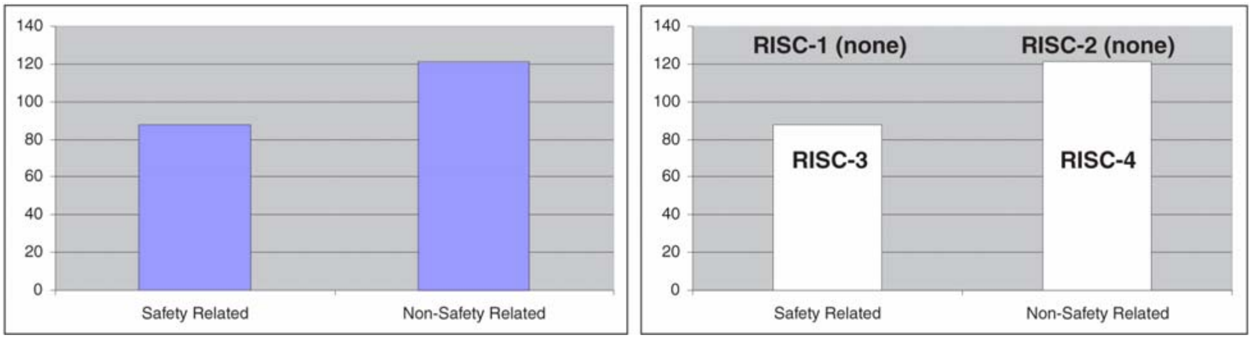


**Figure 3-8**
**Control Building Ventilation System Passive SSCs Categorized Under 10CFR50.69**

### 3.4.7 Charging System Preliminary Active SSC Categorization Results from Surry

Figure 3-9 illustrates preliminary categorization results achieved at Surry for active SSCs in the charging system. The figure shows that after risk-informed categorization, only 20% of the safety-related SSCs were categorized as safety significant, whereas 80% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 18% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

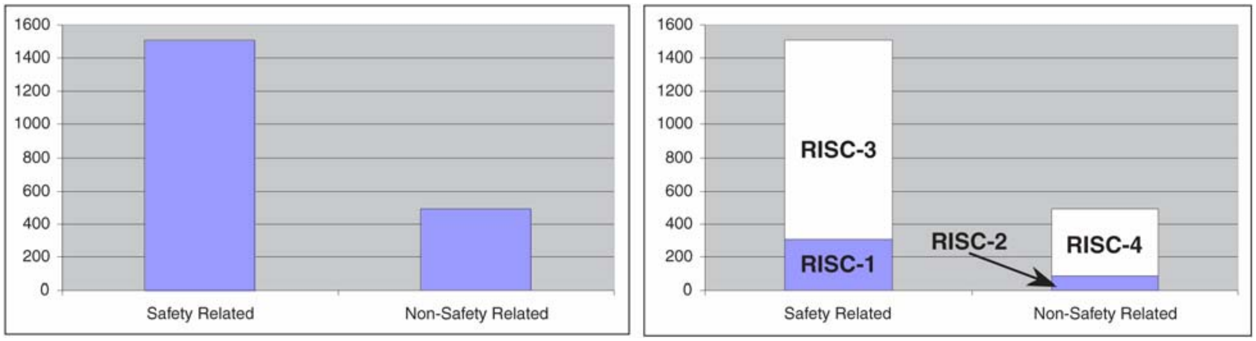


**Figure 3-9**
**Charging System Active SSCs Categorized Under 10CFR50.69**

### 3.4.8 Charging System Preliminary Passive SSC Categorization Results from Surry

Figure 3-10 illustrates preliminary categorization results achieved at Surry for passive SSCs in the charging system. The figure shows that after risk-informed categorization, only 26% of the safety-related SSCs were categorized as safety significant, whereas 74% of those safety-related SSCs were categorized as low safety significant. The figure also shows that only 7% of the originally classified non-safety-related SSCs were subsequently categorized as safety significant.

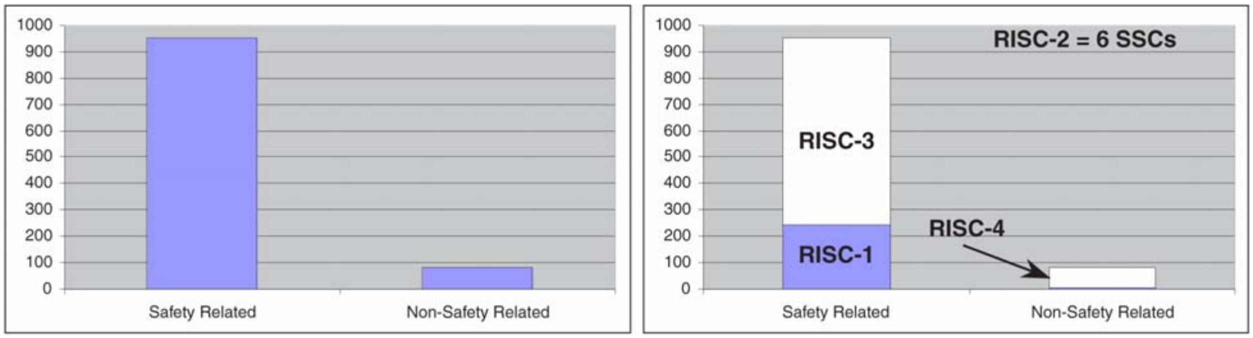


**Figure 3-10**
**Charging System Passive SSCs Categorized Under 10CFR50.69**

### 3.4.9 Charging System Categorization Results from STP and Surry

Figures 3-11 and 3-12 illustrate categorization results achieved at STP and Surry for SSCs in their respective chemical and volume control (charging) systems. The figure shows that after risk-informed categorization, the percentage of the safety-related SSCs that were categorized as safety significant ranged from 8% to 22%. The figure also shows that the percentage of the originally classified non-safety-related SSCs that were subsequently categorized as safety significant ranged from none to 16%.

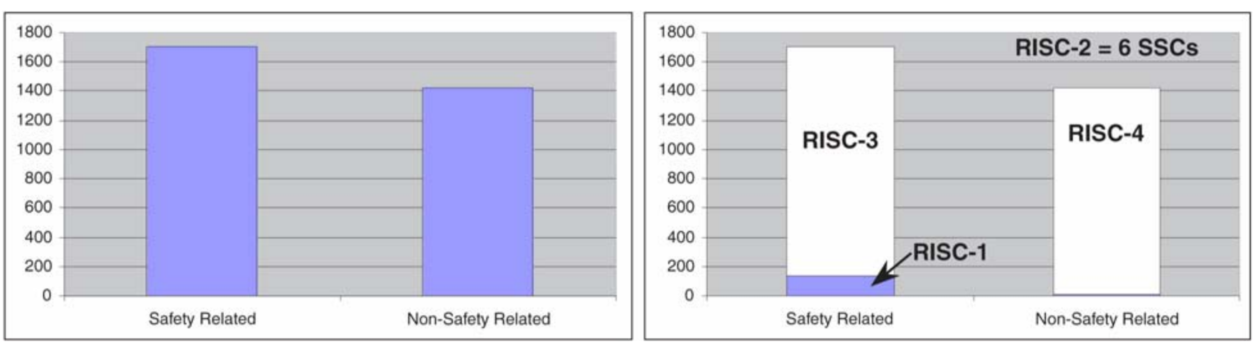


**Figure 3-11**
**STP Chemical and Volume Control System SSCs Categorized Under 10CFR50.69**

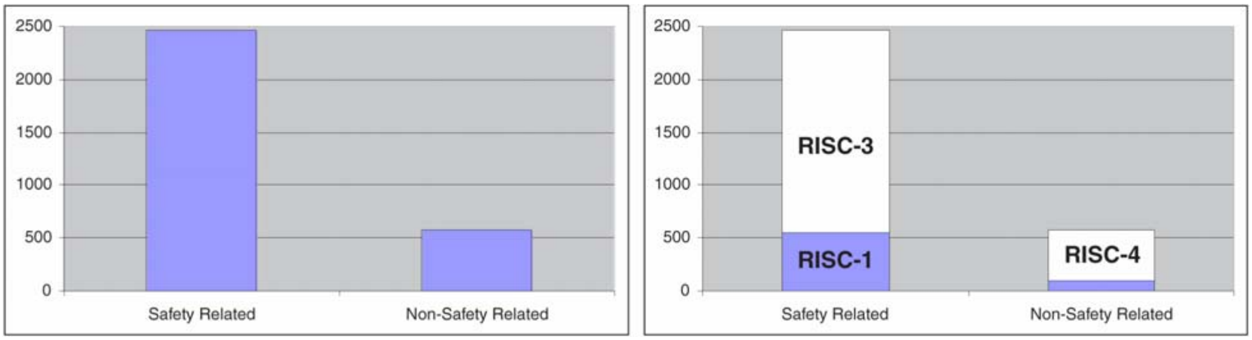


**Figure 3-12**
**Surry Charging System SSCs Categorized Under 10CFR50.69**

The comparative categorizations between STP and Surry suggest that the resulting categorization quantities and percentages are similar in magnitude. Also, the results indicate that even though STP has an extra train, the numbers achieved by the other plant were similar, which suggests that the categorization used by each respective IDP was valid and implemented effectively.

### *3.4.10 Containment Spray Categorization Results from STP and Wolf Creek*

Figures 3-13 and 3-14 illustrate categorization results achieved at STP and Wolf Creek for SSCs in their respective containment spray systems. The figure shows that after risk-informed categorization, the percentage of the safety-related SSCs that were categorized as safety significant ranged from 0% to 2%. The figure also shows that there were none of the originally classified non-safety-related SSCs that were subsequently categorized as safety significant in either categorization scenario.

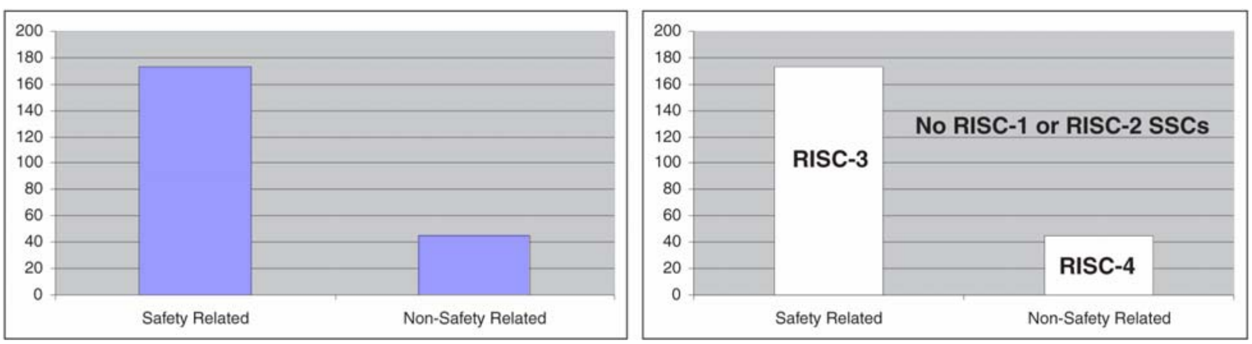


**Figure 3-13**
**STP Containment Spray System SSCs Categorized Under 10CFR50.69**

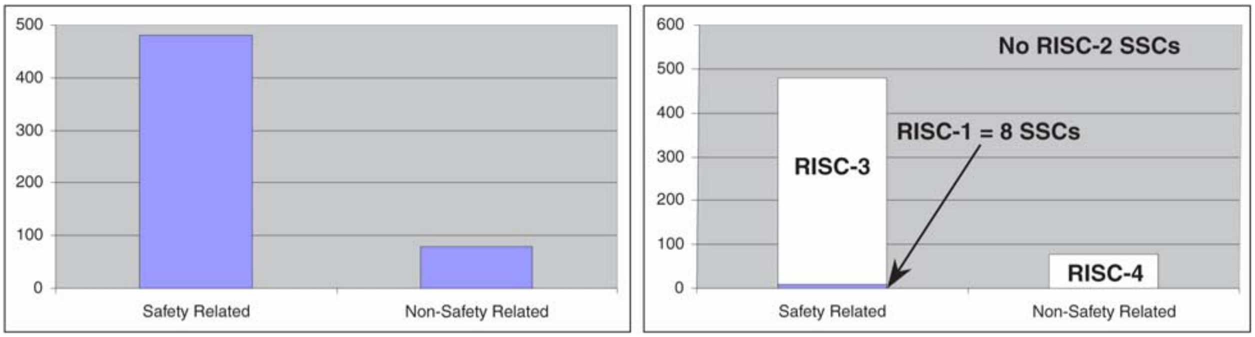


**Figure 3-14**
**WCNOC Containment Spray System SSCs Categorized Under 10CFR50.69**

The comparative categorizations between STP and Wolf Creek suggest that the resulting categorization quantities and percentages are similar in magnitude. Also, the results indicate that even though STP has an extra train, the numbers achieved by the other plant were similar, which suggests that the categorization used by each respective IDP was valid and implemented effectively.

# 4
# ACHIEVING REASONABLE CONFIDENCE

## 4.1 Terminology Used in 10CFR50.69

The term "reasonable confidence" is used three times in 10CFR50.69 as noted below (emphasis added). With regard to the categorization of SSCs, 10CFR50.69(c)(1)(iv) states:

> 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 permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

With regard to the treatment of RISC-3 SSCs, 10CFR50.69(d)(2) states:

> 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.

And with regard to corrective actions for RISC-3 SSCs, 10CFR50.69(d)(2)(iii) states:

> 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.

Although the terminology is used in the regulation, the rule does not define the terms nor make any attempt to describe a measurable or quantifiable level of confidence that would be appropriate. Omission of a definition or clarification leaves the term subjective and open to interpretation by each licensee opting to implement the regulation.

Therefore, the following guidance is provided to assist licensees who have chosen to implement 10CFR50.69 in terms of more clearly understanding the concept of reasonable confidence.

## 4.2 Defining Reasonable Confidence

For the purposes of this report, the term “reasonable confidence” is defined as:

> A level of confidence based on facts, actions, knowledge, experience, and/or observations, which is deemed to be adequate.

The term “actions” in the above definition may constitute verifications, calibrations, tests, or maintenance activities. This is consistent with its use in 10CFR50.69.

Also, for the purposes of this report, the term “reasonable assurance” is used to denote the appropriate and qualitative level necessary for safety-related, safety-significant (RISC-1) equipment performance. This is consistent with its use in 10CFR50 Appendix B, ANSI N45.2 implementing standards, and EPRI guidance cited in this report.

## 4.3 Reasonable Assurance and Reasonable Confidence

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 the health and safety of the public, thereby providing ***reasonable assurance*** of adequate protection to public health and safety. As noted in the preceding paragraphs 10CFR50.69(c)(iv) requires that a licensee or applicant provide ***reasonable confidence*** that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential changes in core damage frequency (CDF) and large early release frequency (LERF) resulting from the implementation of 10CFR50.69 are small.

Text supporting the final rule states that because of the low safety significance of the RISC-3 and RISC-4 SSCs, the additional ***assurance*** provided by qualification testing (or engineering analyses) is not considered necessary.

EPRI Report 102260, *Supplemental Guidance for the Application of EPRI Report NP-5652,* defines reasonable assurance as follows:

> A justifiable level of confidence based on objective and measurable facts, actions, or observations, which infer adequacy.

EPRI Report 102260 also suggests that reasonable assurance can be achieved by using sound engineering judgment. For items that had been deterministically classified as safety related and subsequently controlled under 10CFR50 Appendix B, reasonable assurance of the item’s performance is achieved through the implementation of the 18 criteria contained in the regulation as amplified with applicable ANSI N45.2 standards and plant procedures. EPRI Report 102260 stresses that the assurance gained from all of the processes noted below in Figure 4-1 contributes to achieving an overall level of assurance (that is, reasonable assurance) that the item will perform its design functions.

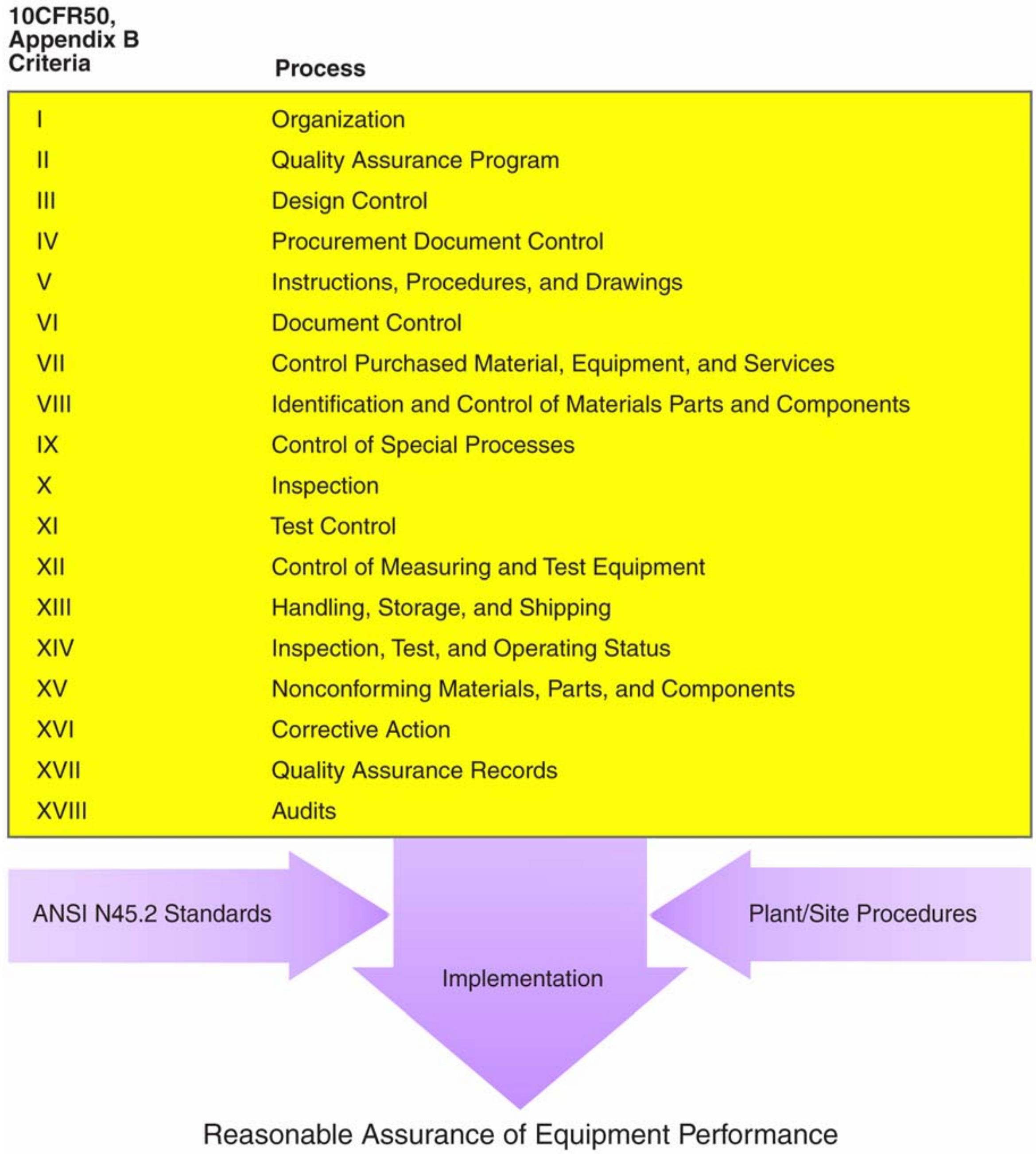


**Figure 4-1**
**Achieving Reasonable Assurance That an Item Will Perform Safety Functions (Based on 10CFR50 Appendix B and EPRI TR-102260)**

## 4.4 Achieving Reasonable Confidence

The user of this report and licensees who have chosen to implement 10CFR50.69 should recognize that the term “reasonable confidence” used within the context of the regulation is still subjective in nature and will vary. Neither this report nor the regulation makes any attempt to quantify the level of confidence that would be deemed appropriate. The resulting level of confidence achieved for RISC-3 items will be different than and less than the reasonable assurance achieved for RISC-1 items, but will still be adequate for the risk significance of the item.

The requirements in 10CFR50.69 can be considered to be performance-based because they give licensees the flexibility to implement treatment that they have determined is needed, commensurate with the low safety significance of certain SSCs in order to provide reasonable confidence that their safety-related functional capability is maintained. In this context, “reasonable confidence” is a somewhat reduced level of confidence as compared with the relatively high level of confidence provided by the current special treatment requirements. Thus alternative treatment for RISC-3 SSCs represents a relaxation of those special treatment requirements applicable for safety-significant SSCs.

Reasonable confidence should be defined and achieved within the context of each particular process (or set of work processes) being implemented. Reasonable confidence can be achieved by considering the following aspects of a work process within a given area of concern (for example, design, configuration management, procurement, maintenance, testing, corrective action, etc.), as shown in Figure 4-2:

- What activities are required within the work process
- How often those activities are performed
- How those activities should be performed/implemented
- The extent to which the implementation of completed processes is documented

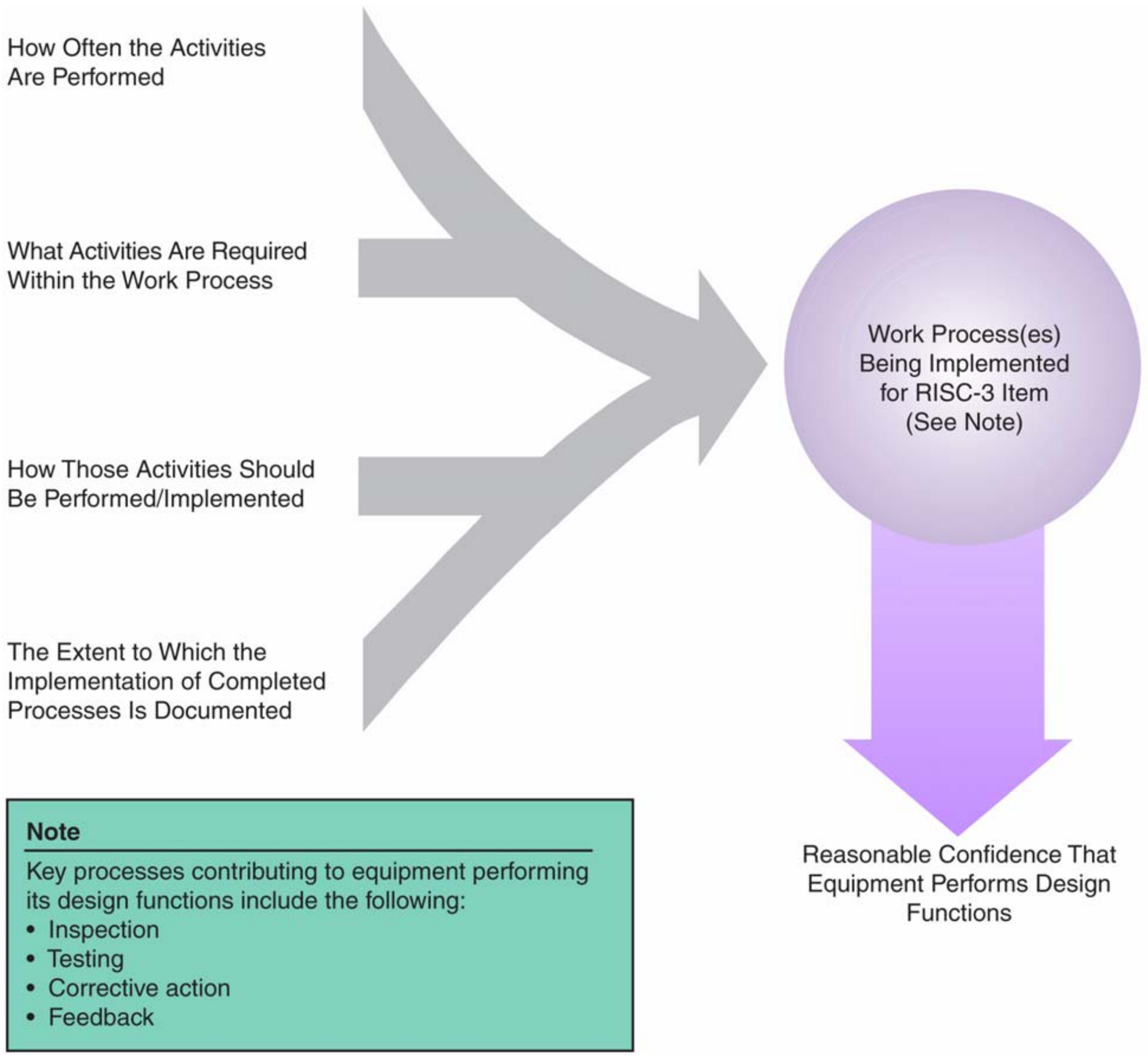


**Figure 4-2**
**Achieving a Reasonable Confidence That a RISC-3 Item Will Perform Design Functions**

Figure 4-3 illustrates a comparison between achieving reasonable assurance and achieving reasonable confidence as those terms are used in the context of this report.

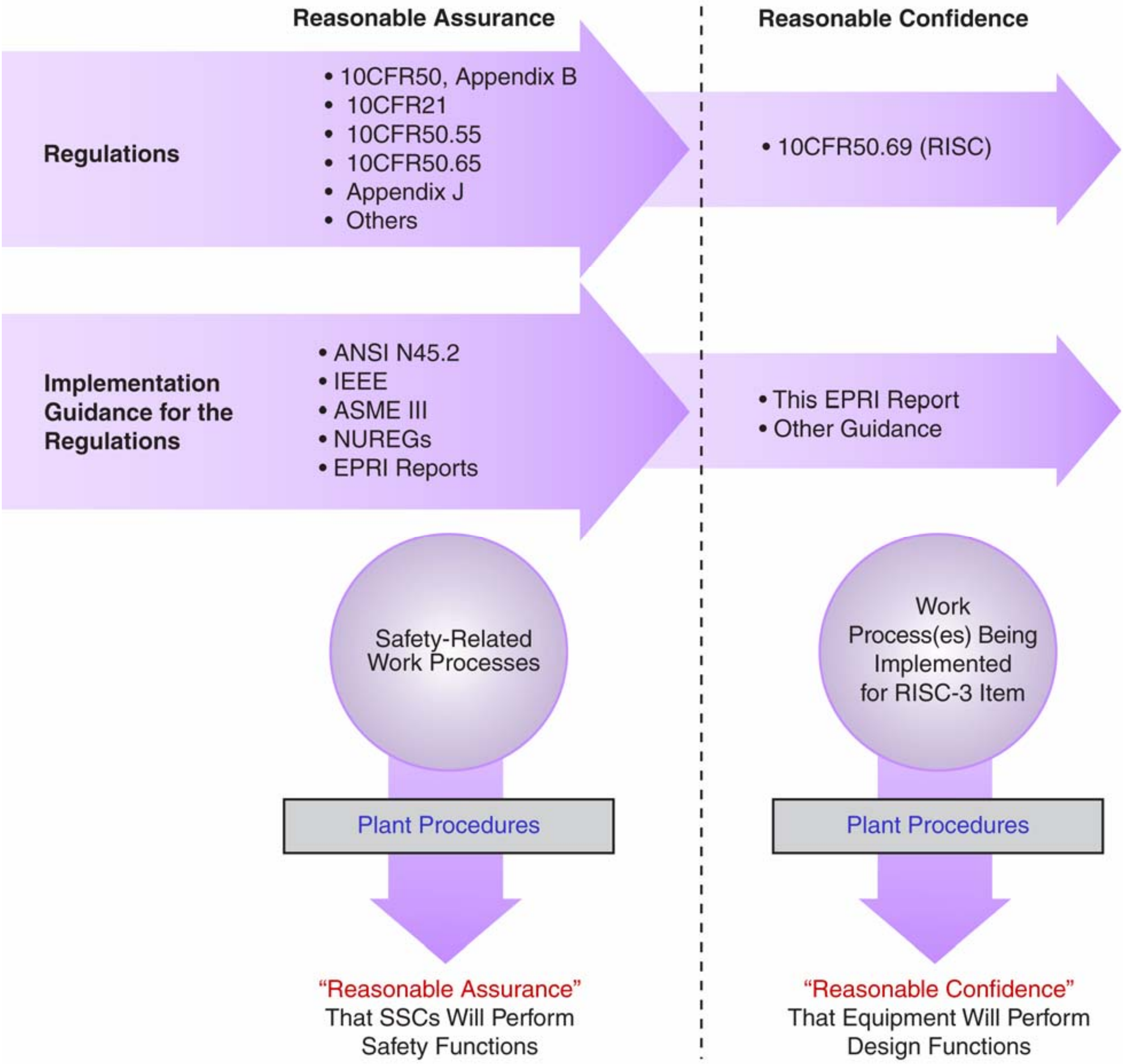


**Figure 4-3**
**Comparing the Terms Reasonable Assurance and Reasonable Confidence**

Each licensee should establish procedures that address these factors for the treatment of structures, systems, and components that have been categorized as RISC-3. Recommended process activities, implementation guidance, and suggested levels of documentation are provided in Sections 6 through 9 of this report and in companion guides EPRI 1008748, *Guidance for Accident Function Assessment for RISC-3 Applications*, and EPRI 1011783, *RISC-3 Seismic Assessment Guidelines*. Examples of how a licensee can apply the guidance to certain work processes and activities are provided in the appendices of this report. In general, the guidance and examples provided here enable the licensee to establish reasonable confidence, within the context of the process or program affected by 10CFR50.69 implementation, that those affected SSCs will continue to perform design functions.

# 5
# DESIGN BASIS INFORMATION

The purpose of this section is to provide licensees with an understanding of design basis concepts and applications at their facility in support of adopting 10CFR50.69.

## 5.1 Reviewing Changes to the Plant

10CFR50.69 does **not** change the design of the plant. Any changes to the design of SSCs are controlled under existing processes, as appropriate. Treatment is established and performed under the basic assumption that the existing design of the categorized item was and remains suitable for its current application(s).

### *5.1.1 Design Bases and Licensing Basis*

The NRC and the Nuclear Energy Institute (NEI) have developed guidance on the definition of design basis. Section C of Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR50.2 Design Bases," states that NEI 97-04 Appendix B, "Guidelines for Identifying 10CFR50.2 Design Bases," (dated November 27, 2000) provides guidance and examples that are acceptable to the staff for providing a clear understanding of what constitutes design bases information. This section discusses the NRC and NEI guidance on design bases and how it relates to RISC-3 components and accident function assessment for RISC-3 components.

#### 5.1.1.1 Design Basis Concepts from NEI 97-04 Appendix B

NEI 97-04 Appendix B (November 2000) provides the 10CFR50.2 definition of design bases as:

> Design bases means "that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals."

NEI 97-04 Appendix B provides general guidance that 10CFR50.2 design bases consist of the following:

- Design bases functions: Functions performed by systems, structures, and components (SSCs) that are (1) required by, or are otherwise necessary to comply with, regulations, license conditions, orders, or Technical Specifications, or (2) credited in licensee safety analyses to meet NRC requirements.
- Design bases values: Values or ranges of values of controlling parameters established as reference bounds for design to meet design bases functional requirements. These values may be (1) established by NRC requirement, (2) derived from or confirmed by safety analyses, or (3) chosen by the licensee from an applicable code, standard, or guidance document.

NEI 97-04 Appendix B continues with the following specific guidance:

- 10CFR50.2 design bases include the bounding conditions under which SSCs must perform design bases functions. These bounding conditions may be derived from normal operation or any accident or events for which SSCs are required to function, including anticipated operational occurrences, design basis accidents, external events, natural phenomena, and other events specifically addressed in the regulations such as station blackout (SBO) and anticipated transient without scram (ATWS).
- The 10CFR50.2 design bases of a facility are a subset of the current licensing basis and are required pursuant to 10CFR50.34(a)(3)(ii) and (b) and 10CFR50.71(e) to be included in the updated FSAR.
- Underlying 10CFR50.2 design bases is substantial supporting design information. Supporting design information includes other design inputs, design analyses, and design output documents. Supporting design information may be contained in the UFSAR (as design description) or other documents, some of which are docketed and some of which are retained by the licensee.

10CFR50.2 design bases functional requirements are derived primarily from the principal design criteria for an individual facility and NRC regulations such as the emergency core cooling system, SBO, and ATWS rules that impose functional requirements or limits on plant design. 10CFR50.2 design bases are a subset of a plant’s licensing basis. While a plant’s licensing basis includes all applicable requirements of Part 50, not all Part 50 requirements have corresponding 10CFR50.2 design bases. For example, in Appendix A, several generic design criteria (GDC) contain requirements for fabrication, construction, testing, inspection, and quality. These are process requirements on SSCs (not requirements for the performance of intended SSC functions) and are, therefore, not 10CFR50.2 design bases.

A significant distinction occurs for 10CRF50.69 RISC-3 components with respect to design bases: 10CFR50 Appendix B does not apply. Therefore, while the description of design bases under 10CFR50.2 and NEI-97-04 Appendix B apply, the requirement for 10CFR50 Appendix B controls of these design bases does not apply. Therefore, the design bases for RISC-3 components may be developed and controlled without applying 10CFR50 Appendix B quality assurance requirements.

### 5.1.1.2 Design Basis Determination for RISC-3

For RISC-3 SSCs, 10CFR50.69 imposes requirements that are intended to retain their design basis capability. Although RISC-3 SSCs are not significant contributors to plant safety, retention of RISC-3 design basis functionality contributes to ensuring that defense in depth and safety margins are maintained. Thus, a newly designed or procured replacement RISC-3 item must be capable of meeting its design basis requirements, even though the special treatment requirements that previously existed are no longer required.

### 5.1.2.3 Application of NEI 97-04 Appendix B Guidance

NEI developed guidelines within NEI 97-04 for determining the boundaries between design basis, licensing basis, supporting design information, and UFSAR information. The basic guideline resulting categorization can best be described in Figure 5-1 below, which depicts the design basis commitments to be a subset of the information within the UFSAR and separate from supporting design information. NEI 97-04 Appendix B gives specific guidance in clarifying the distinction between the design bases and the supporting design information.

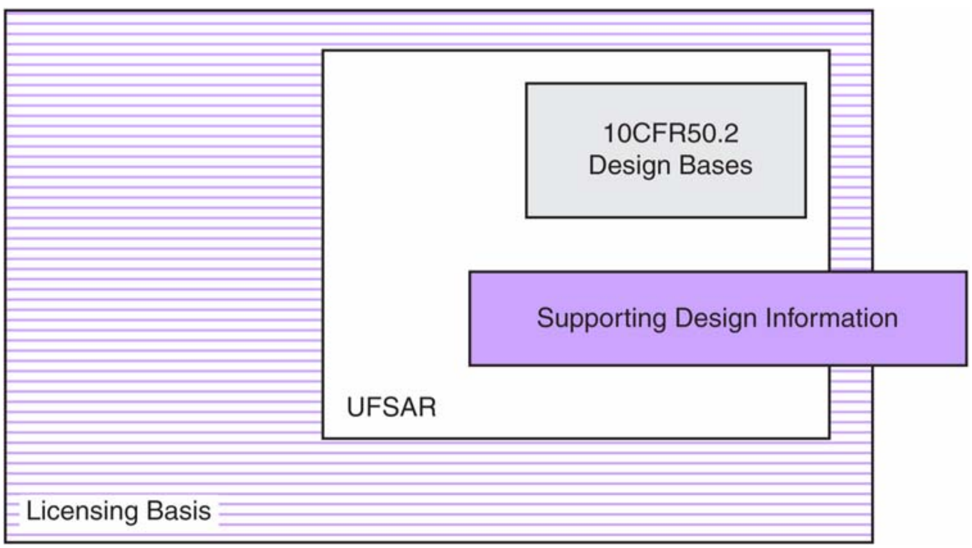


**Figure 5-1**
**Design Basis Determination**

In terms of seismic topical design bases, 10CFR50.2 defines design basis functional requirements as:

> Structures, systems, and components important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. (GDC 2, Design Bases for Protection Against Natural Phenomena, and 10 CFR Part 100, Appendix A)

Similarly, in terms of the tornado topical design bases, 10CFR50.2 defines design basis functional requirements as:

> Structures, systems, and components important to safety shall be designed to withstand the effects of tornadoes without loss of capability to perform their safety function. (GDC 2, Design Bases for Protection Against Natural Phenomena, and 10 CFR 100)

Likewise in terms of the single failure topical design bases, 10CFR50.2 defines design basis functional requirements as:

> Fluid and electrical systems required to perform their intended safety function in the event of a single failure shall be designed to include sufficient redundancy and independence such that neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions. (GDC 17, Electrical power systems; GDC 21, Protection system reliability and testability; GDC 24, Separation of protection and control systems; GDC 25, Protection system requirements for reactivity control malfunctions; GDC 34, Residual heat removal; GDC 35, Emergency core cooling; GDC 38, Containment heat removal; GDC 41, Containment atmosphere cleanup; GDC 44, Cooling water; GDC 54, Piping systems penetrating containment; GDC 55, Reactor coolant pressure boundary penetrating containment; GDC 56, Primary containment isolation)

The user of this report should conclude that while licensing bases and the UFSAR may change as a result of adopting 10CFR50.69, the 10CFR50.2 design bases of the plant do not change.

## 5.2 Design Modifications

### *5.2.1 Changes to Design Bases – General Guidance*

10CFR50.69 does **not** change the design of the plant. Implementation is performed under the basic assumption that the existing design of the categorized item is suitable for its current application(s). For example, the design provisions of each licensee’s construction code of record required by §50.55a(c), (d), and (e) for RISC-3 SSCs are not affected by the rule. Additional guidance regarding treatment of Class 1, 2, and 3 pressure boundary components may be found in related Code Cases.

Another example is a requirement for fracture toughness of particular materials that is part of a licensee’s design requirements; such a requirement would continue to apply when repair or replacement of affected components is undertaken. Licensees are still required by §50.59 to evaluate proposed modifications to design requirements for safety-related SSCs, including those categorized as RISC-3.

### 5.2.2 Design Modification of RISC-1 and RISC-3 SSCs

Any design modification of RISC-1 and RISC-3 SSCs will be controlled under existing processes that meet the requirements of 10CFR50.59 and associated implementing guidance. In general, design modification should address safety-significant SSCs to reduce vulnerabilities and risk. When processing a design modification, it should be assessed from a categorization standpoint; if the categorization changes as a result of the design modification, then subsequent treatment may also require adjustment. Even if the categorization does not change as a result of the design modification, then treatment may still need to be assessed given the scope and nature of the design modification.

The licensee should apply appropriate treatment that is commensurate with the recategorization that may have resulted from the design modification. For example, a design modification could result in a RISC-1 item being recategorized as RISC-3. In this case, the subsequent treatment (design verification activities) would likely change, and design verification activities would no longer be required to satisfy ANSI N45.2.11. The key lesson learned is to involve the IDP early in the design change process so that new system functions/failure mechanisms are addressed, and components are correctly categorized prior to design change approval and procurement of the components required for the design change package (DCP).

## 5.3 Design Control Measures for Components Qualified to Withstand Severe Environments

For RISC-3 SSCs, the rule removes the regulatory requirements for environmental qualification of electric equipment specified in § 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.” However, the rule does not eliminate the requirements in 10CFR50 Appendix A, “General Design Criteria for Nuclear Power Plants,” that electric equipment important to safety be capable of performing its intended functions under the applicable environmental conditions. If RISC-3 electrical equipment is relied on to perform a safety-related function beyond its original design life, licensees should have a basis that justifies the continued capability of the equipment under adverse environmental conditions.

Additional guidance regarding alternative treatment to environmental qualification is provided in EPRI Report 1008748, *Guidance for Accident Function Assessment for RISC-3 Applications (Alternate Treatment to Environmental Qualification for RISC-3 Applications)*.

### 5.3.1 Design Control Measures for Components Qualified to Withstand Seismic Events

RISC-3 and RISC-4 SSCs are required to function under design basis seismic conditions, but they are not required to be qualified by testing or specific engineering methods in accordance with the requirements stated in 10CFR Part 100 Appendix A. A licensee who adopts 10CFR50.69 is no longer required to meet certain requirements in 10CFR Part 100 Appendix A, Sections VI(a)(1) and VI(a)(2) for RISC-3 SSCs, to the extent that those requirements have been interpreted as mandating qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the safe shutdown earthquake and operating basis earthquake. The rule does not remove the design requirements related to the capability of RISC-3 SSCs to remain functional considering safe shutdown earthquake and operating basis earthquake seismic loads, including applicable concurrent loads. These continue to be part of the design basis requirements or procurement requirements for replacement SSCs.

The rule does not change the design input earthquake loads (magnitude of the loads and number of events) used in the design of RISC-3 SSCs. For example, for the replacement of an existing safety-related SSC that has been categorized as RISC-3, the same seismic design loads would still apply. The rule does permit licensees to select a technically defensible method to show that RISC-3 SSCs will remain functional when subject to design earthquake loads. The level of confidence for the design basis capability of RISC-3 SSCs, including seismic capability, should be less than the confidence in the design basis capability of RISC-1 SSCs. The use of earthquake experience data is allowable as a potential method to demonstrate that SSCs will remain functional during earthquakes.

Additional guidance regarding alternative treatment to seismic qualification is provided in EPRI Report 1011783, *RISC-3 Seismic Assessment Guidelines*.

### *5.3.2 Repair/Replacement of Components and Subcomponents*

#### 5.3.2.1 Replacing Components and Subcomponents

When components and subcomponents require replacement, identical items should be used whenever possible. When this is not possible and an alternative item is available, any physical differences between the installed item and the replacement item should be assessed for required design changes or impacts to categorization of SSCs. Depending on the significance of the physical changes occurring with the item, either an equivalent change or design modification may result. Additional guidance regarding how the adoption of 10CFR50.69 impacts the licensee's existing procurement processes is provided in Appendix B of this report.

#### 5.3.2.2 Repair and Replacement of ASME Code Components and Subcomponents

In the Statement of Considerations portion of 10CFR50.69, Section III.4.3 entitled "Section 50.55a(f), (g), and (h) Codes and Standards" states the following:

> Section 50.55a(g) incorporates by reference provisions of the ASME Code as endorsed by NRC that contains the inservice inspection, and repair and replacement requirements for ASME Class 2 and Class 3 SSCs. The Commission will not remove the repair and replacement provisions of the ASME BPV Code required by §50.55a(g) for ASME Class 1 SSCs, even if they were categorized as RISC-3, because those SSCs constitute principal fission product barriers as part of the reactor coolant system or containment. For Class 2 and 3 SSCs that are shown to be of low safety-significance if categorized as RISC-3, the additional assurance from the specific provisions of the ASME Code is not considered necessary.

As such, for Class 2 and Class 3 ASME components that become categorized as RISC-3, the licensee has the opportunity to use alternative treatments during repair and replacement activities. An example of alternative treatment guidance can be found in ASME Code Case N-662, "Alternative Repair/Replacement Requirements for Items Classified in Accordance with Risk-Informed Processes." This ASME Code Case was in development at the time of publication of this report.

In general terms, the licensee can pursue one of the following two options with regard to alternative treatment options during the repair/replacement of Class 2 and Class 3 ASME Code components and subcomponents:

- The licensee can continue to specify and use the existing technical requirements inherent to their current plant code of record. This may be ASME III or, for older plants, ANSI B31.1. This option ensures that the design basis is maintained by continuing to use the same material standards, design margins, testing/inspection requirements, etc., required by their plant code of record. However, the licensee can opt to decrease, or eliminate in some cases, the quality requirements associated with the ASME code or their current plant code of record. Examples of quality assurance requirements that may be decreased or eliminated are:
  - Supplier N-stamp
  - Inspection by an Authorized Nuclear Inspector (ANI)
  - Material traceability
  - Certain documentation provided by the raw material manufacturer or the part manufacturer
- The licensee can specify and use technical requirements inherent to an alternative design standard. This may be ANSI B31.1, ASTM, or AWA in lieu of ASME III. If the alternative design standard allows for the use of different materials, the licensee may also be required to evaluate those alternative materials using the technical equivalency evaluation as described in Section 7 of this report. However, similar to the first option described in the preceding paragraph, the licensee can still opt to decrease, or eliminate in some cases, the quality requirements associated with the ASME code or their current plant code of record.

Primarily, the licensee needs to ensure that the functionality of the component will be maintained by meeting one of the following:

- Construction codes or standards applicable to the items: ASME, ANSI, AWS, AISC, AWWA, API-650, API-620, MSS-SPs, TEMA, and those standards referenced within these documents
- The requirements of NCA, NB, NC, ND, and NF as applicable, but substituting an Owner's Quality Assurance Program (NCA-8140) to verify supplied material conformance in lieu of a Material Organization's Quality System Program (NCA-3800)

# 6
# ESTABLISHING APPROPRIATE TREATMENT

## 6.1 Implementation Overview

Figure 6-1 provides an overview of the implementation phases and summary treatment guidance for each RISC category resulting from the categorization process described in NEI-00-04.

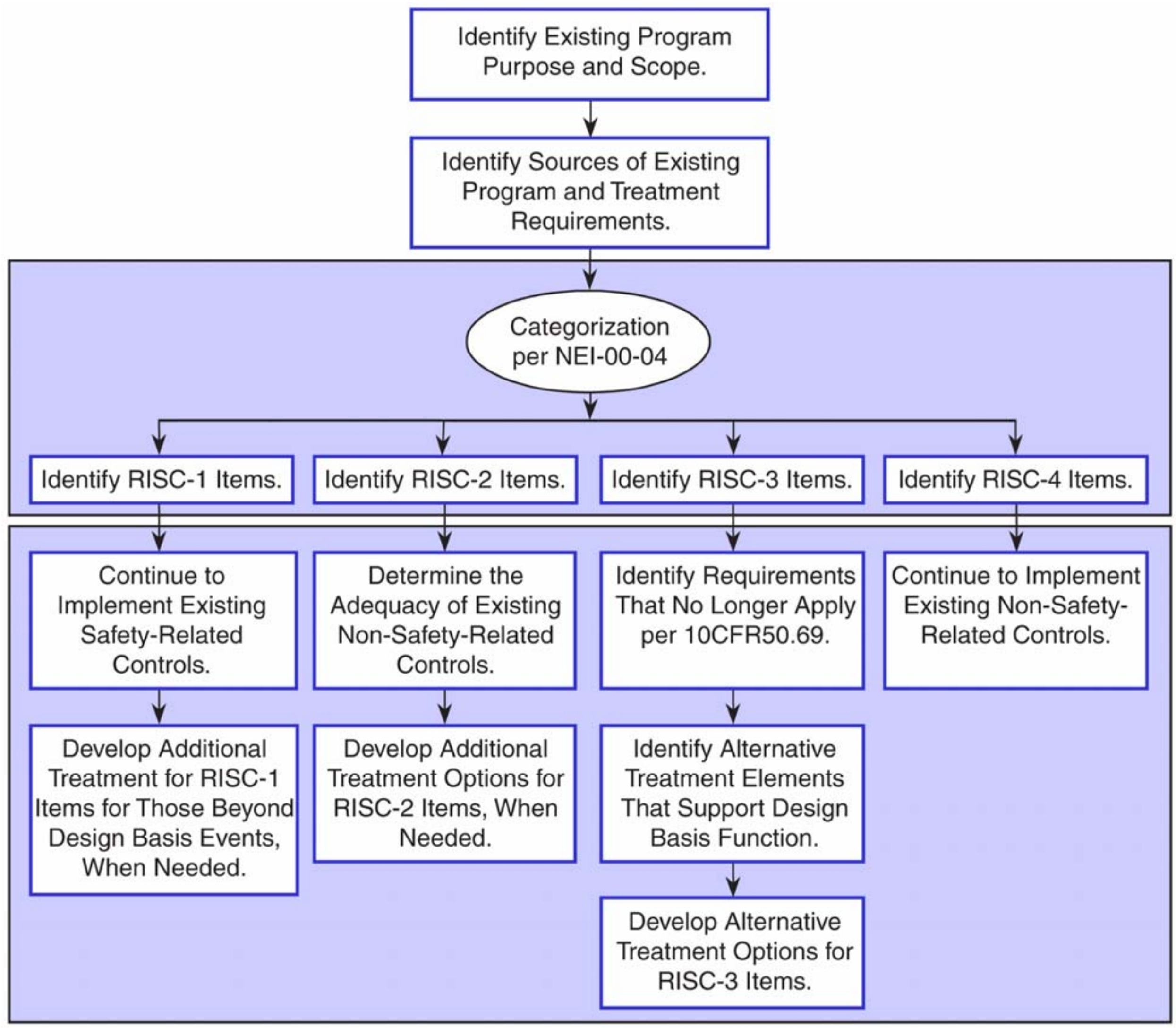


**Figure 6-1**
**10CFR50.69 Implementation Phases and Summary Treatment Guidance**

Figure 6-1 illustrates the following key points:

- The categorization process is separate from the treatment of items under 10CFR50.69. This report assumes that the licensee has already categorized SSCs in accordance with the requirements of NEI 00-04 and that the categorizations are technically accurate.
- For RISC-1 items, the licensee should maintain the existing special treatment. The licensee is responsible, however, for determining if additional requirements are necessary for these SSCs to ensure that their performance remains consistent with the assumed performance in the categorization process (including the PRA) for beyond design basis functions. Because these items have already been classified safety related, licensees should anticipate that only a relatively few items will require additional controls over the existing special treatment.
- For RISC-2 items, the licensee is responsible for determining that existing treatment is adequate for those attributes that made the item risk significant. Experience suggests that licensees should anticipate that only a relatively few items will require alternative treatment to supplement those controls already in place.
- For RISC-3 items, the licensee should identify program requirements that no longer apply, identify program treatment elements that support design basis, and then develop appropriate alternative treatment options for these RISC-3 items.
- No additional treatment considerations or actions are required for items categorized as RISC-4, and in some cases, augmented quality controls may no longer be necessary.

## 6.2 Special Treatment of RISC-1 SSCs

The following guidance is offered in the rule concerning the special treatment of RISC-1 SSCs:

> RISC-1 and RISC-2 SSCs. The licensee or applicant shall ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

RISC-1 SSCs should continue to satisfy all of the existing regulatory requirements that are applicable, including those insights that were considered during the SSC's categorization.

## 6.3 Treatment of RISC-2 SSCs

As noted above, for non-safety-related SSCs determined by the IDP to be safety significant (that is, RISC-2 SSCs), §50.69 maintains the current regulatory requirements (that is, it does not remove any requirements from these SSCs) for special treatment. However, in most cases, regulatory requirements are minimal for these non-safety-related SSCs. Licensees should determine that these current requirements are adequate for addressing design basis performance of these SSCs and that their performance remains consistent with the assumed performance in the categorization process (including the PRA). Ensuring that RISC-2 SSCs continue to perform their functions does not obligate the licensee to add new programmatic controls.

The process for evaluating the treatment of RISC-2 SSCs involves the following actions:

- Identify the attribute(s) that resulted in the subject SSC being categorized as safety significant. The treatment assessment should focus only on ensuring that these critical attributes are satisfied.
- Review the adequacy of the existing treatment (current processes and controls) to ensure that the critical attributes are satisfied. This review will include the equipment history, current performance, maintenance and operational practices, and administrative requirements currently in place.
- Following review, if the current controls are deemed to be adequate to ensure that the critical attribute(s) will be satisfied, no additional treatment controls (inspection, testing, etc.) are required. However, the licensee should document the reasonable confidence basis that the critical attributes will be satisfied and should ensure that the RISC-2 SSC is included in the scope of the Maintenance Rule program (10CFR50.65) and is also subject to the Corrective Action Program.
- Following review, if the current controls are deemed to be inadequate to ensure that the critical attribute(s) will be satisfied, then the licensee must focus on the specific actions needed to provide reasonable confidence that the critical attribute(s) will be satisfied. Example actions to take could include replacing existing hardware that has shown poor performance over the component's history, bolstering preventive maintenance scope and/or frequency to address past component failure histories, performing periodic tests or checks to ensure the continued availability of the component, performing inspections (quality or other) to ensure the adequacy of the work performed or the state of component readiness, etc.
  - In addition to the above treatment controls, the component is to be included in the scope of the Maintenance Rule program and is also to be subject to the licensee's Corrective Action Program.
  - The licensee should document, based on the additional controls applied, the basis for reasonable confidence that the critical attribute(s) will be satisfied.

It is important to note that RISC-2 SSCs do not change their safety classification and are not intended to be placed under full safety-related regulatory controls. Licensees are only to ensure that appropriate controls are placed on the RISC-2 SSC so that reasonable confidence exists that the critical attributes will be satisfied when called upon.

Examples of RISC-2 SSCs could include components such as instrument air compressors, filters, and storage tanks; main feedwater regulating valves and bypass valves; fire protection HVAC dampers; and auxiliary diesel engines.

Appendix B of this report provides examples of how components categorized according to 10CFR50.69 and NEI 00-04 may be subsequently treated to achieve reasonable confidence of safety-significant functionality under 10CFR50.69.

## 6.4 Alternative Treatment of RISC-3 SSCs

### *6.4.1 Requirements of 10CFR50.69*

The following guidance is offered in the rule concerning the alternative treatment of RISC-3 SSCs:

> 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 process. Inspection and testing, and corrective action shall be provided for RISC-3 SSCs.

The licensee should recognize that verification activities encompass both inspection and testing. The rule does not imply that both are necessary in all circumstances.

### 6.4.2 Generic Process for Developing Treatment for RISC-3 Items

Figure 6-2 illustrates the generic process used to develop the treatment provided in Appendix A of this report.

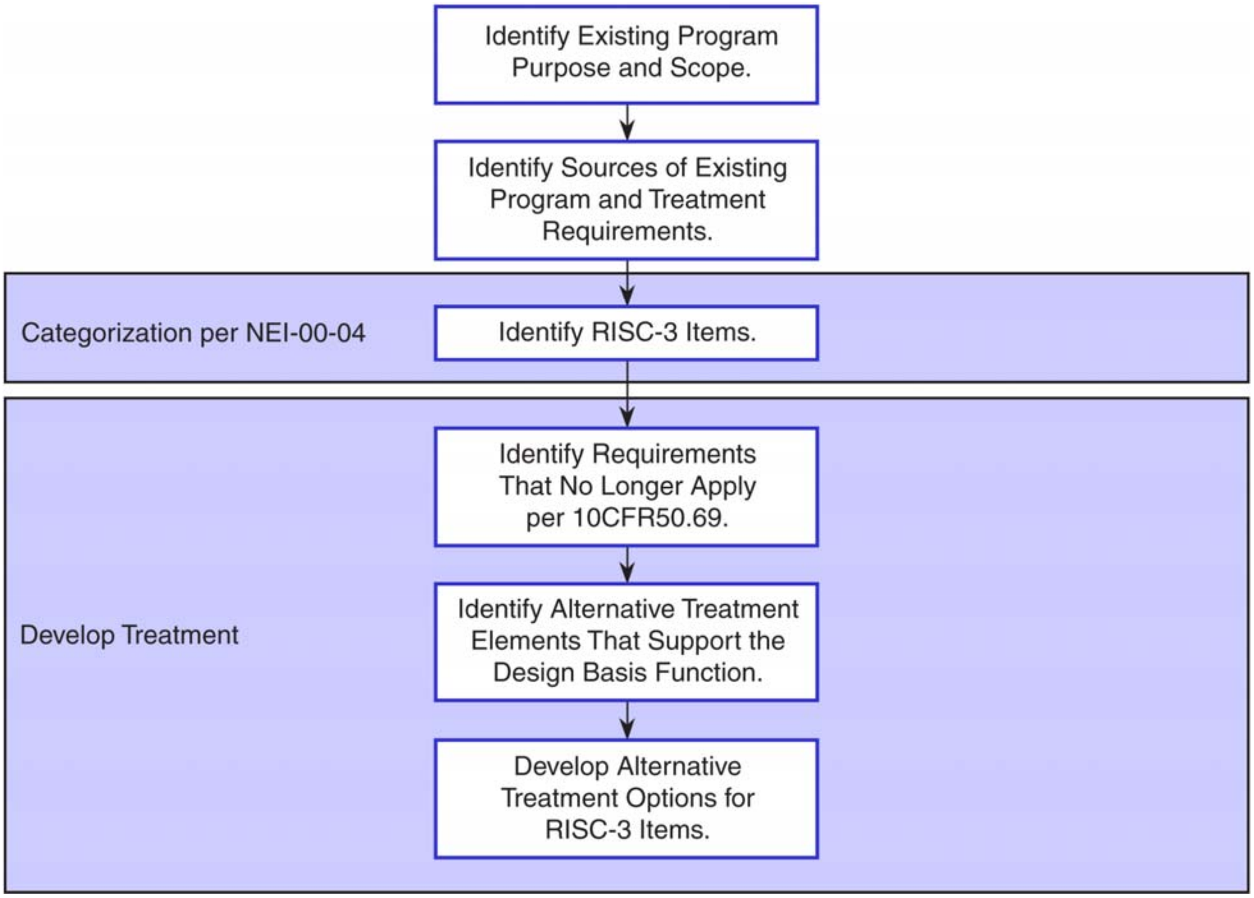


**Figure 6-2**
**Generic Process for Developing RISC-3 Treatment Guidance**

The process illustrated above was used to ensure consistency among the recommendations made in this report and with the guidance cross-referenced from other EPRI reports providing RISC-3 treatment. A more detailed explanation of how this process was used to develop the guidance in this report and its companion reports is provided below.

#### 6.4.2.1 Identify Existing Program Purpose and Scope

The first step in the process is to identify a particular program's purpose and scope. The purpose of this step is to understand the "why" behind the program in a generic sense and to fully understand the reasons why an engineering/maintenance program is undertaken.

Examples are as follows:

- Seismic Qualification Program – Plant response to an earthquake.
- Environmental Qualification Program – Plant response to a LOCA.
- Procurement Processes – Ensure correct items are specified, procured, and accepted in a timely basis to support plant maintenance activities.
- ASME Repair and Replacement – Ensures that repairs and replacements remain in accordance with the construction code of record.
- Quality Assurance Program – Comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality Assurance includes quality control, which comprises those QA actions related to the physical characteristics of a material, structure, component, or system that provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

### 6.4.2.2 Identify Sources of Existing Program and Treatment Requirements

The next step in the generic process is to identify sources of existing program requirements. In a vast majority of cases, the existing sources of programmatic requirements are driven from regulations removed by 10CFR50.69 for low safety-significant SSCs. However, identifying the sources of existing program requirements provides a fuller understanding of the basis for each program being examined.

Examples are as follows:

- Seismic Qualification Program – Part 100 Appendix A, General Design Criteria (GDC) 2, Regulatory Guide 1.100, IEEE 344, SQUG, GIP, plant-specific licensing commitments
- Environmental Qualification Program – 10CFR50.49; IEEE 323
- Procurement Processes – 10CFR50 Appendix B, 10CFR21 Revision 2, ANSI N45.2, ANSI N18.7, EPRI NP-5652 as endorsed by NRC Generic Letter 89-02, Generic Letter 91-05
- ASME Repair and Replacement – ASME Section XI, AWS, ASME Section IX
- Quality Assurance Program – 10CFR50 Appendix B, ANSI N45.2, ANSI/ASME NQA-1, ANS 3.2

### 6.4.2.3 Identify Program Requirements That No Longer Apply per 10CFR50.69

10CFR50.69 eliminates many special treatment requirements for RISC-3 applications. However, at the same time, it defines requirements to be applied to RISC-3 applications and for obtaining “reasonable confidence” in the performance of accident function(s).

10CFR50.69 eliminates the following requirements for applications that have been designated as RISC-3:

- 10CFR50 Appendix B (quality requirements)
- 10CFR Part 21 (deficiency reporting requirements)
- 10CFR50.49 (testing, documentation, and margin requirements for EQ purposes)
- 10CFR50.55(e) (event reporting)
- 10CFR50.55a(f), (g), and (h) (applicable portions of ASME and IEEE codes and standards)
- 10CFR50.65 (Maintenance Rule)
- 10CFR50.72 (notification requirements)
- 10CFR50.73 (licensee event reports)
- Portions of 10CFR50 Appendix J (primary reactor containment leakage testing)
- Portions of Appendix A to 10CFR Part 100 (seismic qualification with respect to extent of testing and types of analyses)

#### 6.4.2.4 Identify Alternative Treatment Elements That Support Design Basis

In this step, the licensee should identify those aspects of the program that have or could have a direct effect on the ability of RISC-3 SSCs to perform their design functions. Treatment options should then be developed for these relevant programmatic elements.

#### 6.4.2.5 Develop Alternative Treatment Options for RISC-3 Items

Given an understanding of the reasons for the program, then appropriate treatment should be developed. The existing programmatic requirements, and in some cases corresponding acceptance criteria, should be used as a benchmark for establishing the new alternative treatment requirements for RISC-3 items. This treatment is described either in Section 7 of this report or companion reports that address a particular engineering/maintenance program encompassed by one of the three major areas referenced in 10CFR50.69 (inspection, testing, and corrective action).

A licensee opting to implement the treatment guidance provided in these reports will achieve reasonable confidence that those SSCs categorized as low safety significant will continue to satisfy design functional requirements.

### 6.4.3 Consideration of SSC Functions During or After an Accident

Another key consideration when establishing appropriate treatment for RISC-3 SSCs is their respective functions during or after an accident. SSCs (components) must be capable of functioning under the applicable accident environment whether new or aged. 10CFR50.69 does not imply that the EQ alternative treatment itself must cover or include all of the inspection and testing requirements for RISC-3 components however. Consideration of inspection, testing, and corrective action should primarily be addressed by the overall plant activities for RISC-3 components, not just during the development of the accident function assessment. This is because the development of an accident function assessment for a RISC-3 component may or may not result in all of the requirements for inspection, testing, or corrective action.

## 6.5 Treatment of RISC-4 SSCs

There are no specific requirements offered in the rule concerning the alternative treatment of RISC-4 SSCs. The rule, however, does not impose any new treatment requirements on RISC-4 SSCs. Instead, RISC-4 SSCs are simply removed from the scope of any applicable special treatment requirements identified in § 50.69(b)(1). Requirements applicable to RISC-4 SSCs not removed by § 50.69(b)(1) continue to apply. Any changes (beyond changes to special treatment requirements) must be made according to existing design change control requirements including § 50.59, as applicable.

# 7
# INSPECTION, TESTING, AND CORRECTIVE ACTION

## 7.1 10CFR50.69 Requirements

10CFR50.69(d)(2) states in full:

> Inspection and testing, and corrective action shall be provided for RISC-3 SSCs.
>
> (i) Inspection and testing. 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; and
>
> (ii) Corrective action. 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.

## 7.2 Inspection and Testing Guidance

### *7.2.1 General Programmatic Guidance*

The rule emphasizes that periodic inspection and testing should continue to be conducted to some extent to provide reasonable confidence that the RISC-3 SSCs will remain capable of performing their safety-related functions. The extent of continued inspection and testing constitutes alternative treatment for these SSCs. Section 6 of this report provides a generic process that licensees can use to determine the most appropriate level of treatment to achieve reasonable confidence.

A number of programs and external sources of input that define periodic inspection and testing requirements already exist at nuclear power plants. Some of them are listed here:

- Local leak-rate testing (LLRT) – See Appendix A for alternative treatment guidance.
- In-service testing (IST)
- In-service inspections (ISI)
- Motor-operated valve testing (MOVs)
- Air-operated valve testing (AOVs)

- Check valve testing/inspections
- Relief valve testing
- Snubber testing/inspections
- Integrated leak-rate testing (ILRTs)
- Equipment manufacturer recommendations
- Industry best-practice documents

Appendix A of this report provides an example of how alternative treatment can be developed regarding local leak-rate testing (LLRT). This report will be revised and amended with other appendices as alternative treatment guidance is developed for the other programmatic areas listed above.

### 7.2.2 Installation and Post-Installation Testing

Licensees are responsible for determining appropriate installation and post-installation testing of RISC-3 SSCs as part of design control and other treatment processes to provide reasonable confidence in the capability of SSCs to perform their functions. Each licensee should continue to control special processes associated with installation, such as heat treatment or NDE, to provide reasonable confidence in the design basis capability of RISC-3 SSCs.

As applicable, licensees may perform post-installation testing to verify that the installed SSC is operating within expected parameters and is capable of performing its safety functions under design basis conditions. When post-installation testing is deemed necessary, licensees may apply engineering analyses to extrapolate the test data to demonstrate design basis capability.

### 7.2.3 Other Inspection and Testing Activities

Under 10CFR50.69, the licensee has latitude in reducing the scope and frequency of other types of inspections and tests inherent to existing work processes. These other types of verification activities include the following:

- CGI dedication inspections/testing (Reference Appendix C of this report)
- Seismic qualification testing (Reference EPRI 1011783, *RISC-3 Seismic Assessment Guideline)*
- EQ testing (Reference EPRI 1008748, *Guidance for Accident Function Assessment for RISC-3 Applications)*
- Equipment functional testing
- In-process manufacturing product inspection/testing
- Final product inspection/testing

## 7.3 Corrective Action

Taking timely corrective action is an essential element for maintaining the validity of the categorization and treatment processes used to implement § 50.69. For safety-significant SSCs, all current requirements continue to apply, and as a consequence, 10CFR50 Appendix B corrective action requirements should continue to be applied to the design basis aspects of RISC-1 SSCs to ensure that conditions unfavorable to quality are corrected. Existing corrective action requirements, if any, for those non-safety items categorized as RISC-2 SSCs would also continue to apply.

When a licensee determines that a RISC-3 SSC does not meet its established criteria for performance of design functions (that is, nonconformances and/or deficiencies regarding the RISC-3 item), the rule requires that a licensee perform timely corrective action (§ 50.69(d)(2)(ii)). Neither the rule nor this report attempts to quantify the timeliness for taking corrective action. However, in most cases, the licensee's existing program for corrective action should be adequate and commensurate with the safety significance of the effected component.

For RISC-3 SSCs, the licensee has the following options regarding corrective action:

- Continue to implement the same corrective action requirements as for other safety-related SSCs (that is, RISC-1 SSCs).
- Develop alternative corrective action requirements applicable only to RISC-3 SSCs. For example, the threshold for entering the corrective action program may be different or the time for taking corrective action may be lengthened.

There are no requirements in the rule for addressing corrective action requirements for RISC-4 SSCs.

# 8
# FEEDBACK AND PROCESS ADJUSTMENT

## 8.1 10CFR50.69 Requirements

With regard to all categories (RISC-1, RISC-2, RISC-3, and RISC-4 SSCs), 10CFR50.69(e)(1) states in full (emphasis added):

> The licensee shall review ***changes to the plant, operational practices, applicable plant and industry operational experience***, and, as appropriate, update the PRA and SSC categorization and treatment processes. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.

With regard to the safety-significant SSCs (RISC-1 and RISC-2 SSCs), 10CFR50.69(e)(2) states in full:

> The licensee shall ***monitor the performance*** of RISC-1 and RISC-2 SSCs. The licensee shall make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid.

With regard to RISC-3 SSCs, 10CFR50.69(e)(3) states in full:

> The licensee shall consider data collected in § 50.69(d)(2)(i) for RISC-3 SSCs to determine if there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations conducted to satisfy § 50.69(c)(1)(iv). The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization process and results are maintained valid.

## 8.2 Reviews of Operational Practices and Operational Experience

The validity of the categorization process relies on ensuring that the performance and condition of SSCs continue to be maintained to be consistent with applicable assumptions. Changes in the level of treatment applied to an SSC might result in changes in the reliability of the SSCs credited in the categorization process. When a nonconformance/deficiency is discovered in the field, the licensee should document the deficiency and assess the need to initiate feedback under 10CFR50.69.

Figure 8-1 illustrates the following key points with regard to handling nonconformances and initiating feedback under 10CFR50.69:

- It is expected that RISC-3 SSCs will periodically degrade and fail (just as RISC-1 SSCs currently under full regulatory treatment controls periodically degrade and fail); however, it is expected that licensees will ensure (with reasonable confidence) that RISC-3 SSCs reliably satisfy their design functional requirements.
- Not all nonconformances necessitate an increase to treatment or a change in categorization.
- Changes to treatment should be considered before changes to categorization.

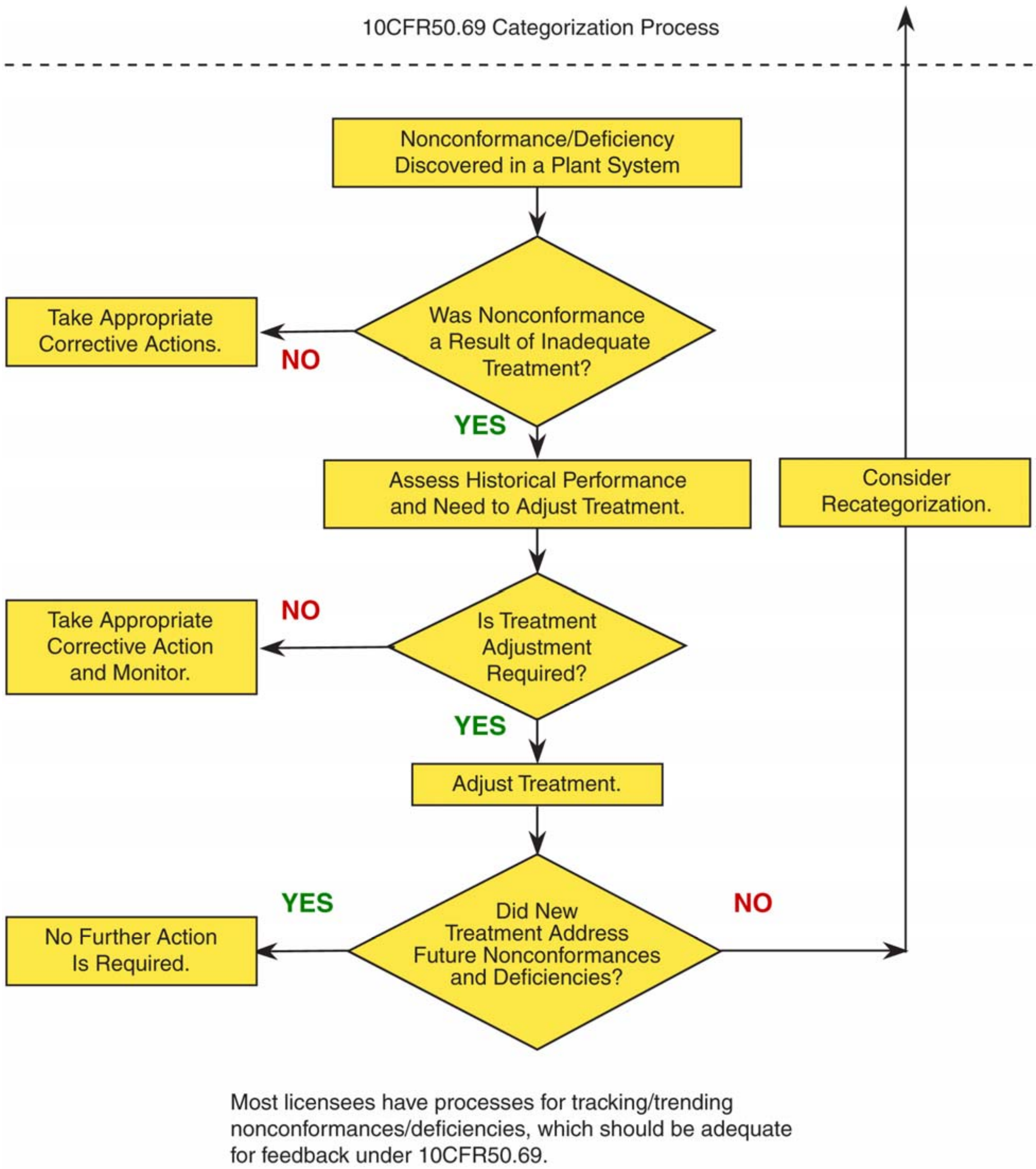


**Figure 8-1**
**Generic Process for Assessing Feedback to RISC-3 Treatment**

As shown in Figure 8-1, when a nonconformance or deficiency is discovered for a RISC-3 component, the licensee should document the deficiency under the Corrective Action program. This documentation appropriately communicates the existence of the deficiency and initiates the needed corrective actions. In addition, the licensee should ask if the deficiency was a result of inadequate treatment. Generally, this question is informally addressed and does not require formal documentation. If the deficiency was not a result of inadequate treatment, the licensee should take appropriate corrective actions to restore the full functionality of the RISC-3 component, and no additional actions are required. If the deficiency was a result of inadequate treatment, the licensee should further assess the historical performance of the failed component and provide additional insight into the need to adjust the existing treatment.

Based on this assessment result, if no treatment adjustment is required, the licensee should correct the deficiency and restore the component to service. In addition, the licensee should monitor the component for further occurrence of this failure mechanism. If the assessment recommends a treatment adjustment, the licensee should implement appropriate adjustments, restore the component to service, and monitor for future failures. The recognition (if the treatment adjustment resolved the deficiency cause) may take a long period of time to conclude. However, at some reasonable point in time, the licensee should be able to conclude if the treatment adjustment was successful.

If the treatment adjustment was successful, the corrective actions in Figure 8-1 are considered completed. If the treatment adjustment was not successful (that is, another failure of the RISC-3 component occurred within a given period of time), then the licensee should consider either further treatment adjustments or a possible categorization change back to RISC-1. A categorization change would reinstitute the full regulatory controls back onto the component. This action should be taken as appropriate, but it should be considered following an adjustment to the treatment controls.

Additionally, plant design changes, changes to the plant-specific PRA model, changes to operational practices, and plant and industry operational experience may impact categorization process results and resulting treatment. Consequently, the rule contains requirements for updating the categorization and treatment processes when conditions warrant in order to ensure that continued SSC performance is consistent with the categorization process and results.

Specifically, the rule requires licensees to review operational practices and applicable plant and industry operational experience, and, as appropriate, to update the PRA and SSC categorization. The review must be performed in a timely manner but at least once every two refueling outages. In addition, licensees are required to obtain sufficient information on SSC performance to verify that the categorization process and its results remain valid. For RISC-1 SSCs, much of this information may be obtained from present programs for inspection, testing, surveillance, and maintenance. However, for RISC-2 SSCs and for RISC-1 SSCs credited for beyond design basis accidents, licensees need to ensure that sufficient information is obtained. For RISC-3 SSCs, there is a relaxation of the requirements for obtaining information when compared to the applicable special treatment requirements. However, sufficient information still needs to be obtained. The rule requires considering performance data, determining if adverse changes in performance have occurred, and making the necessary adjustments so that desired performance is achieved so that the evaluations conducted to meet § 50.69(c)(1)(iv) remain valid. The feedback and adjustment process is crucial to ensuring that the SSC performance is maintained to be consistent with the categorization process and its results.

To accomplish the required “periodic review,” a licensee should proactively schedule a formal review to occur within the once per two refueling outages period specified in the Rule. The Corrective Action program maintained by the licensee is a valuable source of information to be used during the periodic review process. Any tests or inspection deficiencies, as well as operational concerns or industry operating experience issues, should be captured within the licensee’s Corrective Action program and should be readily retrievable. In addition, the licensee should assess any changes in system health from the appropriate system engineer and should seek any recommendations from the system engineer on potential categorization changes that may be required. Updated insights should be provided on design changes, PRA model changes, licensing document changes, operating experience changes, operational changes, quality changes, or other changes that have occurred since the initial categorization that might have an influence on the current categorization recommendation. These data sources should provide the needed insight to determine if the categorized components continue to satisfy their design functional requirements and continue to satisfy the assumptions in the original categorization bases.

Based on the review performed, it is expected that some components may require an adjustment to their current categorization. The adjusted treatments for these new categorizations are detailed in Section 8.3. In addition, while a thorough periodic review is required to be completed on a once per two refueling outages basis, it is recommended that categorization adjustments be considered on an on-going periodic basis as new insights are raised that may affect the current categorization of a component. This is especially true if a process change (for example, design change, PRA model change, or other changes) could possibly change a component categorization from RISC-3 to RISC-1 or, to a lesser extent, from RISC-4 to RISC-2.

## 8.3 Treatment Adjustments Following Component Categorization Changes

Treatment of SSCs should be reviewed and possibly revised following a change in component categorization. It is not expected that a large number of SSCs will require categorization revision following initial categorization; however, it is expected that some recategorizations will occur that are largely driven by required updates to the plant-specific PRA. Licensees should monitor the PRA revision process closely and based on expected changes to PRA categorization inputs, be prepared to potentially recategorize components and adjust their associated treatment. Treatment should not, however, be an input or driver for the categorization revision. Following a categorization change, treatment should be increased commensurate with any changes in categorization from RISC-3 to RISC-1 and from RISC-4 to RISC-2. Treatment should be decreased commensurate with any changes in categorization from RISC-1 to RISC-3 and from RISC-2 to RISC-4.

Licensees should maintain an electronic repository for noting the correct and most current categorization of SSCs. This may typically be the plant Q-list, but other equipment databases, such as those integral to many commercially-available materials management information systems, would also be feasible. The electronic repository chosen to retain the categorization results should be readily accessible by plant personnel. It is not expected that plant design drawings will be impacted by the categorization process alone. However, due to resulting treatment adjustments to RISC-3 SSCs, some design drawing changes are envisioned (for example, if an ASME Class 3 pipe section is replaced with a design equivalent ANSI B31.1 pipe section). If a licensee chooses to include categorization data on design drawings, changing the categorization of SSCs depicted on plant design drawings should be performed at the discretion of each licensee, based upon the adequacy of their respective equipment databases.

The licensee should consider the following process for addressing scenarios where changes in treatment may be needed following a change in categorization:

- For changes from RISC-1 to RISC-3, licensees should assess the full scope of treatment reductions that could occur for the changed component as was done during the initial categorization process. If certain program and process changes have already occurred to initially implement the 10CFR50.69 treatment allowances, the new RISC-3 component may be added to the scope of the revised program or process and treated accordingly.
- For changes from RISC-3 to RISC-1, it is assumed that the licensee has already implemented treatment reductions (up to full treatment allowances specified within 10CFR50.69) while the component was categorized as RISC-3. Upon recognition that a recategorization to RISC-1 has occurred, the licensee should treat this change as critical and should perform an assessment on the required reinstallment of regulatory controls onto the component in a timely manner. These assessments should consider the following insights:
  - The Control Room (Shift Supervisor) should be notified that a possible operability determination may be required since the new RISC-1 component does not currently satisfy the RISC-1 regulatory treatment requirements.
  - Appropriate surveillance testing may be required to permit the component to be determined operable under the RISC-1 regulatory controls.
  - Inclusion of the component back into the scope of regulatory programs (for example, local leak-rate testing program, Maintenance Rule program, in-service testing program, EQ program, seismic qualification program, etc.) should occur in a timely manner. In addition, the rebaselining of the RISC-1 treatment for this component should be discussed.
  - A review of in-process work packages, parts procurements, etc should occur to ensure that appropriate RISC-1 controls are in place.
  - Updating of the electronic database to reflect the subject component as RISC-1 should occur in a timely manner.
- For changes from RISC-2 to RISC-4, licensees should assess the removal of any regulatory treatment or additional controls that may have been imposed while the component was categorized as RISC-2. Licensees should focus on applying commercial controls only to SSCs determined to be RISC-4.

- For changes from RISC-4 to RISC-2, licensees should assess the component for possible additional controls as was done for new RISC-2 SSCs during initial categorization. These assessments should occur in a timely manner, but it is expected, as previously stated, that generally the commercial controls already will be sufficient to ensure that the critical attributes of the new RISC-2 SSC will be satisfied. However, the following insights should be considered:
  - Historical performance of the component while under previous commercial controls.
  - Operating experience focused on the critical attribute(s) of the component. If the operating experience shows no concerns with satisfying the critical attribute, this aids in concluding that existing commercial controls may be adequate.
  - Determining if additional programs/processes exist that would add value in ensuring that the critical attribute(s) is satisfied.
  - Including the new RISC-2 SSC in the scope of the Maintenance Rule, if it is not already included.
  - Ensuring that the new RISC-2 SSC is properly included in the scope of the Corrective Action program.

# 9
# ADMINISTRATIVE CONTROLS

## 9.1 10CFR50.69 Requirements

10CFR50.69(f) contains administrative requirements for keeping information current, for handling planned changes to programs and processes, and for maintaining records. Each subparagraph is discussed below. The rule states in full:

> (1) The licensee or applicant shall document the basis for its categorization of any SSC under paragraph (c) of this section before removing any requirements under § 50.69(b)(1) for those SSCs.
>
> (2) Following implementation of this section, licensees and applicants shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with § 50.71(e).
>
> (3) When a licensee first implements this section for a SSC, changes to the FSAR for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.59 evaluation of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met.
>
> (4) When a licensee first implements this section for a SSC, changes to the quality assurance plan for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.54(a) review of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the quality assurance plan may be made if the requirements of this section and § 50.54(a) continue to be met.

## 9.2 Program Documentation

10CFR50.69(f)(1) states that the licensee or applicant shall document the basis for categorization of SSCs in accordance with this section before removing any requirements. The documentation is expected to address why a component was determined to be either safety significant or low safety significant, based upon the requirements in § 50.69(c). Documentation of the technical bases for categorization should be performed in accordance with the guidance contained in NEI 00-04. As such, this report will contain no further discussion of this requirement.

## 9.3 Change Control

10CFR50.69(f)(2) specifies that the licensee must update its UFSAR to reflect which systems have been categorized using the provisions of § 50.69 and, thus, may have revised treatment applied to the structures and components within those systems. This provision is included to maintain clear information, at a minimum level of detail, about which requirements a licensee is satisfying; detailed information about particular SSCs is not required to be submitted. For an applicant, this updating would be expected to be either part of the original application or a supplement to the UFSAR under § 50.34. For licensees, the updating must be in accordance with the provisions of § 50.71(e).

Once the NRC has approved the licensee's § 50.69 submittal, the licensee may proceed with categorization and treatment adjustment. NRC does not plan to perform a pre-implementation review of the revised treatment requirements under § 50.69(d). However, the Commission recognizes that existing information in the quality assurance (QA) plan or in the UFSAR may need to be revised to reflect the changes to treatment that would be made as a result of implementation of § 50.69. Any revisions to these documents are to be submitted in accordance with the existing requirements of § 50.54(a)(2) and § 50.71(e), respectively. For instance, § 50.71(e) states that the UFSAR is to contain the latest information developed and is to reflect information submitted to the Commission since the last update. Appropriate sections of the rule further state how a licensee should submit revisions to the QA plan or the UFSAR. Information in these documents that would no longer be accurate upon implementation of § 50.69 must be updated. Details of the processes would be expected to be contained in plant procedures, procurement documents, surveillance records, and other documents.

10CFR50.69(f)(3) specifies that for initial implementation of the rule, changes to the UFSAR for implementation of the rule need not include a supporting § 50.59 evaluation of changes directly related to implementation. This is true since the License Amendment Request submitted by the licensee to voluntarily adopt 10CFR50.69 includes a complete and thorough review by the NRC prior to approval. This review and approval by the NRC includes both the categorization approach that will be followed and the high-level treatment adjustments that will be applied commensurate with the component's categorization. Therefore, in granting an amendment to a licensee's license, the NRC has granted prior approval of the implementation changes that need to be reflected in the UFSAR. So, during initial implementation of the approved NRC allowances, no additional § 50.59 evaluation is required for changes to the UFSAR since the NRC has already approved the License change. However, future changes to the treatment processes and procedures for § 50.69 implementation may be made, provided the requirements of the Rule and § 50.59 continue to be met. Although 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 changes to these parts of the UFSAR.

10CFR50.69(f)(4) specifies that for initial implementation of the rule, changes to the quality assurance plan for implementation of the rule need not include a supporting § 50.54(a) review of changes directly related to implementation. Similar to the reasoning provided above for 10CFR50.69(f)(3), based on the thorough NRC review given during the initial 10CFR50.69 approval process, no additional NRC review is necessary to initially implement the needed changes to the quality assurance plan. Therefore, a supporting § 50.54(a) review is not required for initial implementation. However, future changes to the treatment processes and procedures for § 50.69 implementation may also be made, provided the requirements of the rule and § 50.54(a) continue to be met. Although the licensee is to update its programs to reflect implementation of § 50.69, the Commission concluded that no additional review under § 50.54(a) is necessary for changes to these parts of the QA plan.

## 9.4 Records Retention

No explicit requirements are included in § 50.69 for the period for retention of records. The rule specifies only a few specific types of records that must be prepared, for example, those for the basis for current categorization in § 50.69(f)(1). In accordance with § 50.71(c), these types of records are to be maintained until the Commission terminates the facility's license.

## 9.5 Reporting

10CFR50.69(g) provides a new reporting requirement applicable to events or conditions that would have prevented a RISC-1 or RISC-2 SSC from performing a safety-significant function. The rule states in part:

> The licensee shall submit a licensee event report under § 50.73(b) for any event or condition 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 § 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 assumptions made in the categorization process. This reporting requirement is intended to capture these situations. The reporting requirement is contained in § 50.69, rather than as a revision of § 50.73, so that its applicability only to those facilities that have implemented § 50.69 is clear.

The existing reporting requirements in § 50.72 and § 50.73 no longer apply to RISC-3 (and RISC-4) SSCs under the rule.

# 10
# STRATEGIC GUIDANCE FOR ADOPTING 10CFR50.69

The purpose of this section is to provide licensees with some strategic guidance if they are contemplating the adoption of 10CFR50.69 based on the experiences and lessons learned from South Texas Project and the pilot plants.

## 10.1 Project Initiation Strategy

Adoption of 10CFR50.69 involves a significant, forward-looking strategic decision for a licensee. While the benefits of enhanced nuclear safety and improved operating efficiency appear to be a win-win situation, 10CFR50.69 does involve a change to how the nuclear plant is licensed. A change to the license is a sizeable undertaking and—once committed to—is not easily undone. The types of changes that accompany 10CFR50.69 implementation are far-reaching and can touch virtually all aspects of a nuclear organization. Therefore, buy-in and communication are key elements when initially determining to undertake a 10CFR50.69 implementation approach.

After a decision is made to adopt 10CFR50.69, a licensee must determine the degree of categorization and implementation that will be pursued. Much of this insight will be factored into the License Amendment Request (LAR) that the licensee submits to the NRC for approval. However, the decision on how much to categorize largely determines the level of resources that will be required to complete and maintain the categorization output. In addition, the amount of categorization completed will also dictate the amount of benefit to be gained from 10CFR50.69. It is important to note that only categorized components receive potential benefits from 10CFR50.69 implementation; if a component is not categorized, it will retain its current regulatory treatment until categorization is completed.

Upon deciding how many systems to categorize and what those systems are, the licensee will focus its energy on the categorization process. This includes constituting a technically sound integrated decision-making panel (IDP) with appropriate support from the senior management team. The IDP process is covered in NEI-00-04 and will not be detailed here; however, it is emphasized that the integrity of the IDP is of utmost importance in order to appropriately categorize components according to the industry-approved guidelines and to properly document the conclusions and the bases that formed those conclusions. This documentation will be referred to throughout the life of the plant to understand the bases for the initial categorization, to aid in performing categorization adjustments, or to satisfy regulatory inquiries.

In parallel with the categorization activities, the licensee should decide on a few programs or processes that will initially implement the 10CFR50.69 treatment allowances. It is recommended that the number of programs initially implemented be small to ensure that the pace and quality of the implementation activities can be properly controlled and assessed for expected results. It is recommended that the initial programs/processes selected for implementation would have a willing program owner who is supportive and eager to implement the 10CFR50.69 allowances. The selected program/process should also have a fairly narrow focus with a single owner/coordinator to manage the communication and implementation. Some example programs that could be recommended for initial implementation include the Maintenance Rule program and the local leak-rate testing program.

When preparing the chosen program/process for implementation, appropriate procedure revisions are required to remove the RISC-3 SSCs from the regulatory controls. In addition, the alternative RISC-3 treatments must be determined and appropriately applied. These RISC-3 treatments may require the generation of new preventive maintenance (PM) tasks, or the revision of PM frequencies based on the determined RISC-3 treatment. Those organizations affected by the process changes should be included in the implementation planning activities, and necessary training should occur to ensure understanding on why the process changes are occurring and what regulatory allowances exist to support the changes. Active communication is encouraged throughout the implementation process to ensure that those affected organizations offer insights to enhance the revised process.

As the implementation activities continue, an independent assessment should be planned to ensure that the license requirements are being satisfied and that the expected results are achieved. This may be accomplished using in-house resources supplemented by off-site experts. Necessary adjustments to the implementation process should occur early to minimize a significant amount of process adjustment that could occur as a result of the assessment team’s feedback.

As additional confidence is gained in the initial implementation activities, expansion of the implementation to additional programs/processes should be undertaken. Each new area of implementation will require additional communication and the involvement of additional personnel to support the implementation activities.

Implementing 10CFR50.69 is part of a licensee’s continuous process improvement. The culture changes needed for successful implementation occur over a period of time, and a licensee should undertake implementation activities at an appropriate pace to ensure the viability of the categorization and implementation.

## 10.2 System Selection Strategy for Categorization

The adoption and subsequent implementation of 10CFR50.69 at a nuclear power plant is optional and is dependent on the particular systems deemed appropriate to categorize in accordance with NEI 00-04. Adoption does not imply that all systems need to be categorized, and the adoption of the rule can be limited to the degree that the licensee deems appropriate. However, once a system has been selected for component categorization, then all components in that system must be categorized from a risk-informed perspective.

As noted in Section 4 of this report, to date STP has categorized 88 systems, which constitute 75,382 SSCs. Of these, 20,103 are safety related, and 55,279 are non-safety related. After risk-informed categorization, the following results were achieved:

| | |
|---|---|
| RISC-1 | 4772 SSCs (or 23.7% of safety-related SSCs) |
| RISC-2 | 625 (or 1.1% of non-safety-related SSCs) |
| RISC-3 | 15,331 (or 76.3% of safety-related SSCs) |
| RISC-4 | 54,654 (or 98.9% of non-safety-related SSCs) |

It is neither expected nor required for a licensee to accomplish this same degree of component categorization to achieve benefits in enhanced nuclear safety and efficient plant operations. A licensee who voluntarily adopts 10CFR50.69 may choose to categorize only 10 to 15 key systems that constitute a significant portion of the regulatory requirements and treatments. By performing component categorization, the true safety significance of these systems and components will be better understood and will be better managed by both the licensee and the regulator.

Table 10-1 provides a listing of potential categorization results for 15 systems that a PWR licensee may obtain when adopting 10CFR50.69. The 15 listed systems are generally viewed as “important” by licensees and the regulator, and they have a degree of regulatory control imposed upon them. This table is based on actual work and results at the South Texas Project.

**Table 10-1**
**Listing of Candidate Systems for Risk-Informed Categorization**

| System Name | Deterministic Classifications | | Risk-Informed Categorizations | | | |
|---|---|---|---|---|---|---|
| | **SR** | **NSR** | **RISC-1** | **RISC-3** | **RISC-2** | **RISC-4** |
| Auxiliary Feedwater System | 666 | 164 | 212 | 454 | 10 | 154 |
| Component Cooling Water System | 2037 | 626 | 120 | 1917 | 0 | 626 |
| Chilled Water Systems | 1265 | 130 | 264 | 1001 | 0 | 130 |
| Containment Spray System | 173 | 45 | 0 | 173 | 0 | 45 |
| Chemical and Volume Control System | 1708 | 1419 | 140 | 1568 | 6 | 1413 |
| Emergency Diesel Generator System | 1976 | 722 | 213 | 1763 | 0 | 722 |
| Essential Cooling Water System | 976 | 333 | 156 | 820 | 0 | 333 |
| Main Feedwater System | 670 | 2568 | 104 | 566 | 0 | 2568 |
| Electrical Auxiliary Building HVAC System | 1273 | 996 | 529 | 744 | 15 | 981 |
| Main Steam System | 600 | 1436 | 240 | 360 | 0 | 1436 |
| Radiation Monitoring System (Area and Process) | 360 | 1384 | 0 | 360 | 0 | 1384 |
| Reactor Coolant System | 729 | 621 | 482 | 247 | 87 | 534 |
| Residual Heat Removal System | 379 | 90 | 138 | 241 | 18 | 72 |
| Steam Generator Blowdown System | 206 | 1023 | 60 | 146 | 0 | 1023 |
| Safety Injection System | 1073 | 553 | 290 | 783 | 6 | 547 |

# 11
# REFERENCES

## 11.1 Regulations and Regulatory Guidance


10CFR100 Appendix A, *Seismic and Geologic Siting Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR100.11, *Determination of Exclusion Area Low Population Zone and Population Center Distance*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR21 Revision 2, *Reporting of Defects and Noncompliances*, 1995, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50 Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Facilities*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50 Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.2, *Definitions*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.34, *Contents of Applications; Technical Information*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR 50.62, *Requirements for the Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.69, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.71, *Maintenance of Records, Making of Reports*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.72, *Immediate Notification Requirements for Operating Nuclear Power Reactors*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

10CFR50.73, *Licensee Event Report System*, Office of the Federal Register, National Archives and Records Administration, U.S. Government Printing Office, Washington, DC.

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NRC Regulatory Guide 1.174, July 1998, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

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NUREG–1493, "Performance-Based Containment Leak- Test Program," dated September 1995.

SECY–00–0194, "Risk-Informing Special Treatment Requirements," dated September 7, 2000.

SECY–02–0176, "Proposed Rulemaking to Add New Section 10 CFR50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components," dated September 30, 2002.

SECY–04–0109, "Final Rulemaking to Add New Section 10 CFR50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated June 30, 2004.

SECY–98–300, "Options for Risk-Informed Revisions to 10 CFR Part 50—'Domestic Licensing of Production and Utilization Facilities,"' dated December 23, 1998.

SECY–99–256, "Rulemaking Plan for Risk-Informing Special Treatment Requirements," dated October 29, 1999.

## 11.2 Implementing Standards

ANSI N18.7/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," 1976.

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ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants."

ANSI N45.4-1972, "American National Standard Leakage-Rate Testing of Containment Structures for Nuclear Reactors."

ANSI/ANS-56.8-1994, "American National Standard Containment System Leakage Testing Requirements."

IEEE Std 323-1974, *Qualifying Class 1E Equipment for Nuclear-Power Generating Stations, IEEE*, Piscataway, NJ.

IEEE Std 344-1987, *Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*, IEEE, Piscataway, NJ.

## 11.3 Industry Guidance

*10 CFR50.69 SSC Categorization Guideline*, Nuclear Energy Institute, Washington, DC, December 2004, NEI-00-04.

ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair and Replacement." American Society of Mechanical Engineers, New York.

ASME Code Case N-662, "Alternative Repair/Replacement Requirements for Items Classified in Accordance with Risk-Informed Processes." American Society of Mechanical Engineers, New York.

ASME Code Case N-720, "Risk Informed Safety Classification for Construction of Nuclear Facility Components, Section III, Division 1, Subsections NCA, NB, NC, ND, and NF." American Society of Mechanical Engineers, New York.

ASME/ANSI OM Part 10, "Operations and Maintenance Standards for Inservice Testing of Valves in Light-Water Power Plants." American Society of Mechanical Engineers, New York.

ASME Boiler and Pressure Vessel Code - Section XI, IWV-1100 "Valve Testing." American Society of Mechanical Engineers, New York.

*Equipment Reliability Process Description.* INPO AP-913, Revision 1, Atlanta, GA, 2003.

*Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J*, Nuclear Energy Institute, Washington, DC, July 1995, NEI-94-01, Rev. 0.

*Option 2 Implementation Guide*, Nuclear Energy Institute, Washington, DC, December 2000. NEI-00-04 (Draft Revision A2).

## 11.4 EPRI Technical Reports

*Critical Component Identification Process – Licensee Examples (Scoping and Identification of Critical Components in Support of INPO AP-913.* EPRI, Palo Alto, CA: 2003. 1007935.

*Guidance for Accident Function Assessment for RISC-3 Applications.* EPRI, Palo Alto, CA: 2005. 1008748.

*Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants.* EPRI, Palo Alto, CA: 1994. TR-103586, Revision 2.

*Guidelines for Technical Evaluation of Replacement Items in Nuclear Power Plants, Revision 1*. EPRI, Palo Alto, CA: 2004. 1008256.

*Guidelines for Technical Evaluation of Replacement Items in Nuclear Power Plants (NCIG-11).* EPRI, Palo Alto, CA: 1989. NP-6406.

*RISC-3 Seismic Assessment Guidelines.* EPRI, Palo Alto, CA: 2005. 1011783.

*Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals.* EPRI, Palo Alto, CA: August 1994. TR-104285.

*Supplemental Guidance for the Application of EPRI Report NP-5652 on Commercial Grade Items.* EPRI, Palo Alto, CA: 1994. TR-102260.

# A
# ALTERNATIVE RISC-3 TREATMENT EXAMPLE

This appendix provides an example of how the generic process for developing alternative treatment for RISC-3 SSCs (described in Section 6 of this report) can be applied to an existing program involving testing of certain plant components. The testing program selected for this illustrative example was local leak-rate testing (LLRT).

This example is provided for illustrative purposes only. The alternative treatment developed in this example should be used as a benchmark against which licensees can establish treatment that is consistent with plant-specific commitments, programs, and procedures. Based on plant-specific input, treatment may vary from what is shown in this example.

## A.1 Example Scenario

As a Type C LLRT example, a safety-related inside-reactor-containment (IRC) motor-operated valve (MOV) that returns component cooling water flow from the reactor containment fan coolers during normal plant operations was selected. The selected valve is an ASME Class-2, 14-inch (35.6-cm) Rockwell International valve, model 14-L151C1-Z-S6. This valve has a design basis function to remain open to return cooling water from the reactor containment fan coolers and has an additional function to close on a containment isolation signal.

During the categorization of this valve, it was identified that the valve was modeled in the PRA with a low risk ranking. Deterministically, it was identified that the valve's performance has been excellent in the industry, valve disc failure was determined not to be credible, and normal system parameters provided adequate insight into the status of the valve. The IDP categorized this example valve as low safety significant.

## A.2 Existing LLRT Program Purpose and Scope

Appendix J to 10CFR50, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, states the following regarding the purpose and scope of LLRT:

> One of the conditions of all operating licenses for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and

systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

## A.3 Existing LLRT Program and Treatment Requirements

Existing program and treatment requirements for local leak rate testing for most utilities are derived from the following documents:

- 10CFR50 Appendix J (*Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors*)
- NRC Regulatory Guide 1.163, September 1995 ("Performance-Based Containment Leak-Test Program")
- Nuclear Energy Institute, NEI-94-01, Rev. 0, July 26, 1995 (*Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50 Appendix J*)
- ANSI N45.4-1972 ("American National Standard Leakage-Rate Testing of Containment Structures for Nuclear Reactors")
- ANSI/ANS-56.8-1994 ("American National Standard Containment System Leakage Testing Requirements")
- ASME Boiler and Pressure Vessel Code - Section XI, IWV-1100 ("Valve Testing")
- NRC Generic Letter No. 89-04 ("Guidance on Developing Acceptable Inservice Testing Program")
- ASME/ANSI OM Part 10 ("Operations and Maintenance Standards for Inservice Testing of Valves in Light-Water Power Plants")

## A.4 LLRT Program Requirements That No Longer Apply According to 10CFR50.69

### *A.4.1 10CFR Part 50 Appendix J Containment Leakage Testing*

10CFR50.69(b)(1)(x) removes a subset of RISC-3 and RISC-4 SSCs from the scope of the requirements in Appendix J to 10 CFR Part 50 that pertain to containment leakage testing. Specifically, RISC-3 and RISC-4 SSCs that meet specified criteria in § 50.69(b)(1)(x) are removed from the scope of the requirements for Type B and Type C testing. It is important to note that this removes only the Appendix J leakage testing requirements from these SSCs. These SSCs must still be capable of performing their design basis functions (that is, to close or isolate containment). The basis for the removal of the Appendix J leakage testing requirements follows.

One of the conditions of all operating licenses for water-cooled power reactors as specified in § 50.54(o), is that primary reactor containments shall meet the containment leakage test requirements set forth in Appendix J to 10 CFR Part 50. These test requirements provide for pre-operational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components that penetrate containment of water-cooled power reactors and establish the acceptance criteria for these tests. As such, these tests are special treatment requirements. The purposes of the tests are to ensure both of the following:

- Leakage through the primary reactor containment, or through systems and components penetrating primary containment, does not exceed allowable leakage rate values as specified in the Technical Specifications.
- Periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and systems and components penetrating primary containment.

Appendix J includes two options: Option A and Option B. Option A includes prescriptive requirements, and Option B identifies performance-based requirements and criteria for pre-operational and subsequent periodic leakage rate testing. A licensee may choose either option for meeting the requirements of Appendix J.

The discussion contained in Appendix J to 10 CFR Part 50 can be divided into two categories. Parts of Appendix J contain testing requirements. Other parts contain information, such as definitions or clarifications, necessary to explain the testing requirements. A review of Appendix J did not identify any technical requirements other than those describing the methods of the required testing. Therefore, Appendix J is considered to be, in its entirety, a special treatment requirement.

Although the 1995 revision to Appendix J was characterized as risk informed, the changes were not as extensive as those expected by the inclusion of Appendix J within the scope of § 50.69. The 1995 revision to Appendix J primarily decreased testing frequencies, whereas risk-informing the scope of SSCs that are subject to Appendix J testing removes some components from testing (that is, to the extent that defense in depth is maintained in accordance with the risk-informed categorization process).

### A.4.2 Types of Tests Required by Appendix J

Appendix J testing is divided into the following three types:

- Type A tests are intended to measure the primary reactor containment overall integrated leakage rate after the containment has been completed and is ready for operation and at periodic intervals thereafter.
- Type B tests are intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary. Primary reactor containment penetrations that are required to be Type B tested are identified in Appendix J.
- Type C tests are intended to measure containment isolation valve leakage rates. The containment isolation valves (CIVs) that are required to be Type C tested are identified in Appendix J.

### A.4.3 Reduction in Scope for Appendix J Testing

Type A testing: The rule does not change the Type A testing requirements of Appendix J.

Type B testing: The rule does not change the Type B testing requirements for air lock door seals, including door operating mechanism penetrations that are part of the containment pressure boundary and doors with resilient seals or gaskets, except for seal-welded doors. However, the Commission concludes that Type B testing is not necessary for other penetrations that are determined to be of low safety significance and that meet one or both of the following criteria:

1. Penetrations pressurized with the pressure continuously monitored
2. Penetrations less than 1 inch (2.54 cm) in equivalent diameter

Type C testing: The rule concludes that Type C testing is not necessary for valves that are determined to be of low safety significance and that meet one or more of the following criteria:

1. The valve is required to be open under accident conditions to prevent or mitigate core damage events.
2. The valve is normally closed and is in a physically closed, water-filled system.
3. The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary.
4. The valve size is 1-inch (2.54-cm) nominal pipe size or less.

### *A.4.4 Basis for Reduction of Scope*

The first category of penetrations that are excluded from Type B testing is penetrations that are pressurized with the pressures in the penetrations continuously monitored by licensees. This monitoring would detect significant leakage from the penetrations. The monitoring of the pressures in the penetrations, in conjunction with the requirements for RISC-3 SSCs (including taking corrective action when an SSC fails), ensures with reasonable confidence, without the need for Type B testing, that these penetrations are functional.

The second category of penetrations excluded from Type B testing is penetrations that are 1-inch (2.54-cm) nominal size or less. These penetrations do not contribute to large early releases. Accordingly, the failure of such penetrations does not contribute in a significant way to safety or increased risk. The Commission concludes that such penetrations will not be subject to Type B testing.

Regarding Type C containment leakage testing, the Commission finds that for the four categories of containment isolation valves (CIVs) identified in § 50.69(b)(1)(x), the removal of Type C testing requirements is reasonable because even without Type C testing, the probability of significant leakage during an accident (that is, leakage to the extent that public health and safety is affected) is small.

Appendix J to 10 CFR Part 50 deals only with leakage rate testing of the primary reactor containment and its penetrations. It assumes that the CIVs are in their safe position. No failure is assumed that causes the CIVs to be open when they are supposed to be closed. The valve would be open if needed to transmit fluid into or out of containment to mitigate an accident or closed if not needed for this purpose. For purposes of this evaluation, it is assumed that an open valve is capable of being closed. The licensee or applicant implementing § 50.69 must apply treatment to RISC-3 CIVs that ensures with reasonable confidence that those valves are capable of performing their safety-related function to close under design basis conditions. Testing to ensure the capability of CIVs to reach their safe position is not within the scope of Appendix J and, as such, is not within the scope of this evaluation. Therefore, the valves addressed by this evaluation are considered to be closed, but they may be leaking. The increase in risk due to these SSCs being removed from the scope of Appendix J requirements is negligible.

The acceptability of the removal of Appendix J leakage testing for the RISC-3 CIVs is based on the assumption that those valves are capable of achieving the full seated position by means of the actuator. Therefore, even though a RISC-3 CIV might be exempt from Appendix J leakage testing, the RISC-3 CIV must meet the treatment requirements in § 50.69(d) to provide reasonable confidence that the CIV can perform its safety function (for example, to close) under design basis conditions. Because it is likely that most CIVs will be categorized as RISC-3, the licensee or applicant must evaluate the proposed change in the treatment of RISC-3 CIVs to ensure that defense in depth is maintained by ensuring with reasonable confidence that the RISC-3 CIVs are capable of performing their safety-related functions under design basis conditions. Although the licensee or applicant is allowed flexibility in addressing this issue, the rule requires that the licensee or applicant ensures with reasonable confidence the capability of RISC-3 CIVs to perform their safety functions to maintain defense in depth as discussed in RG 1.174.

Past studies (for example, NUREG–1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants; Final Summary Report," dated December 1990) show that the overall reactor accident risks are not sensitive to variations in containment leakage rate. This is because reactor accident risk is dominated by accident scenarios in which the containment either fails or is bypassed. These very low probability scenarios dominate predicted accident risks due to their high consequences.

The Commission examined the effect of containment leakage on risk in more detail as part of the Appendix J to 10 CFR Part 50, Option B, Rulemaking. The results of these studies are applicable to this evaluation. NUREG–1493, "Performance-Based Containment Leak- Test Program," dated September 1995, calculated the containment leakage necessary to cause a significant increase in risk and found that the leakage rate must typically be approximately 100 times the Technical Specification leak rate, $L_a$. It is improbable that even the leakage of multiple valves in the categories under consideration would exceed this amount. Operating experience shows that most measured leaks are much less than 100 times $L_a$.

A more direct estimate of the increase in risk for the revision to Appendix J can be obtained from EPRI Report TR–104285, *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals*, dated August 1994. This report examined the change in the baseline risk (as determined by a plant's individual plant evaluation [IPE] risk assessment) due to extending the leakage rate test intervals. For the pressurized water reactor (PWR) large dry containment examined in the EPRI report, for example, the percent increase in baseline risk from extending the Type C test interval from 2 to 10 years was less than 0.1 percent. While this result was for a test interval of 10 years vs. the current proposal to do no more Type C testing of the subject valves for the life of a plant, the analysis may reasonably apply to this situation because it contains several conservative assumptions that offset the 10-year time interval. These assumptions include the following:

- The study used leakage rate data from operating plants. Any leakage over the plant's administrative leakage limit was considered a leakage failure. An administrative limit is a utility's internal limit and does not imply violation of any Appendix J limits. Therefore, the probability of a leakage failure is overestimated.
- Failure of one valve to meet the administrative limit does not imply that the penetration would leak because containment penetrations typically have redundant isolation valves. While one valve may leak, the other valve may remain leak-tight. The study assumed that failure of one valve in a series failed the penetration. Therefore, the probability of a penetration leak is overestimated.
- The analysis assumed possible leakage of all valves subject to Type C testing, not just those subject to the relief according to § 50.69.

According to this analysis, the removal of SSCs from the scope of Appendix J requirements does not have a significant effect on risk. The NUREG–1493 analysis shows that the amount of leakage necessary to significantly increase risk is two orders of magnitude greater than a typical Technical Specification leakage rate limit. Therefore, the risk to the public will not significantly increase due to the relief from the requirements of Appendix J to 10 CFR Part 50.

## A.5 Regulatory Treatment Elements That Currently Support Design Basis

Regulatory treatment elements that support the design basis validation include the following:

- Types of tests required by 10CFR50 Appendix J
- Frequency of those tests
- Scope of equipment requiring that those tests be performed

## A.6 Alternative Treatment Options for RISC-3 Items

### *A.6.1 Penetrations No Longer Subject to Type B Testing*

In lieu of Type B regulatory testing for RISC-3 penetrations that satisfy one or more of the criteria listed above, a licensee could consider the following alternative treatments:

- When it is noted that a pressurized penetration fails to satisfy its intended pressure range, generate a Condition Report to document the deficiency, and correct it in a timely manner to restore the correct pressure range.
- An appropriate post-maintenance test (PMT) associated with the rework would include a verification that pressure is being held in the correct range and that the monitoring device is functioning as expected.
- Preventive maintenance (PM) tasks could be developed, as appropriate, to periodically check and calibrate the pressure sensing device.

The alternative treatment is intended to provide reasonable confidence that the penetration continues to satisfy its design basis requirement. Additional treatments may be imposed to provide a satisfactory basis of reasonable confidence.

### *A.6.2 Valves No Longer Subject to Type C Testing*

Type C testing is not necessary for valves that are determined to be of low safety significance and that meet one or more of the following criteria:

1. The valve is required to be open under accident conditions to prevent or mitigate core damage events.
2. The valve is normally closed and is in a physically closed, water-filled system.

3. The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary.
4. The valve size is 1-inch (2.54-cm) nominal pipe size or less.

For the example valve provided in Section A.1, this safety-related, low safety significant valve would reside in the RISC-3 categorization box. However, to reduce treatment as specified in Appendix J and as allowed by 10CFR50.69, this valve must satisfy one or more of the screening criteria identified above. This example valve satisfies criterion 3: "The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary." Based on satisfying one or more of the screening criteria, the treatment for this valve can be reduced as allowed by 10CFR50.69.

The example valve has two design basis functions:

- To remain open during normal plant operations to return cooling water flow from the reactor containment fan coolers
- To close on a containment isolation signal

To ensure, with reasonable confidence, that the example valve will satisfy its design basis functions, the following alternative treatments could be imposed:

- Normal plant parameters provide sufficient insight into the valve's ability to remain open. If the valve fails to satisfy this capability, a Condition Report would be generated, and the valve would be reworked in a timely manner to restore this function.
- Following corrective maintenance, a post-maintenance test (PMT) would be performed to ensure that the valve strokes smoothly, will go and remain open, and will close on demand.
- A periodic "lube-and-inspect" PM task would be developed to provide assurance that the valve remains in a condition to perform its function
- Any MOV-related tests or inspections, as appropriate and as dictated by the MOV program, would continue to be imposed.
- If the valve is used as an isolation boundary during normal maintenance activities, and it is noted that the valve leaks by when pressure is retained on one side, then a Condition Report would be generated, and the valve would be reworked in a timely manner. The PMT for the corrective work could include a leak check, up to and including, the performance of an LLRT.

These alternative treatments are intended to provide reasonable confidence that the valve continues to satisfy its design basis requirements. Additional treatments may be imposed to provide a satisfactory basis of reasonable confidence.

# B
# ALTERNATIVE RISC-1 AND RISC-2 TREATMENT EXAMPLES

This appendix provides examples of how components categorized according to 10CFR50.69 and NEI 00-04 may be subsequently treated to achieve reasonable confidence of safety-significant functionality under 10CFR50.69.

These examples are provided for illustrative purposes only. The resulting categorization, though based on actual plant results, may vary at other plants due to plant-specific conditions and the subsequent findings of the respective IDP. Care should be taken not to assume that the categorization results will apply in all plant applications.

The alternative treatment developed in these examples should be used as a benchmark against which licensees can establish treatment that is consistent with plant-specific commitments, programs, and procedures. Again, based on plant-specific input, treatment may vary from what is shown in these examples.

The scope of examples in this appendix is as follows:

B.1 RISC-2 Reactor Recirculation Pump Breakers

B.2 RISC-1 Reactor Coolant System Pressure Operated Relief Valve

B.3 RISC-2 Feedwater Control Logic System

B.4 RISC-2 Fire Protection Pumps

## B.1 Reactor Recirculation Pump Breakers

This example is illustrates the categorization and subsequent treatment for the reactor recirculation pump breakers and related control circuitry. Originally, this item was classified deterministically as non-safety related. After implementing 10CFR50.69, it was categorized as RISC-2. Further evaluation demonstrated that the existing treatment was still adequate.

| SYSTEM: | FUNCTION: | MULTIPLE FUNCTIONS: |
|---|---|---|
| ATWS | Trip reactor recirculation M-G set breaker (active) | No |
| SCENARIO DESCRIPTION:<br>During an anticipated transient without reactor scram (ATWS) scenario, either automatic or operator-actuated (for example, identified in the emergency operating procedures [EOPs]) trips are credited in the plant PRA. This function includes the RRP-MG set motor and generator field breakers and their associated control circuitry.<br>FAILURE MODE: Fail to open breaker<br>FAILURE RATE/PROBABILITY (FR/FP): 1E-02<br>FR/FP BASES: Taken from PRA notebook and plant response to ATWS rule (10CFR50.62, *Requirements for the Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants*) | | |
| FUNCTION TO BE VERIFIED: Breaker opens on demand. | | |
| EXISTING TREATMENT PRACTICES:<br>The plant's response to the ATWS rule resulted in plant modifications and accompanying testing, maintenance, and surveillance procedures to be implemented to ensure reliable performance of the installed non-safety-related equipment. This system is also included in the Maintenance Rule as it performs a safety significant function. Thus, plant practices have been defined to ensure reliable operation during a demand on this equipment. | | |
| 10CFR50.69 TREATMENT PRACTICES:<br>The existing treatment practices provide reasonable confidence that the reactor recirculation pump breakers will reliably trip when called upon as credited in the categorization process. | | |

## B.2 Reactor Coolant System Pressure-Operated Relief Valve

This example illustrates the categorization and subsequent treatment for a pressure-operated relief valve (PORV) in the reactor coolant system. Originally, this item was classified deterministically as safety related because the pressure-boundary (passive) function was originally considered safety related, but the active functions were originally considered non-safety related. After implementing 10CFR50.69, it was categorized as RISC-1. Further evaluation demonstrated that the existing treatment was still adequate.

### *B.2.1 Evaluation for Bleed Path Function*

| SYSTEM: | FUNCTION: | MULTIPLE FUNCTIONS: |
|---|---|---|
| Reactor Coolant (RC) | Provide the bleed path during feed and bleed cooling (active) | Yes |

| |
|---|
| SCENARIO DESCRIPTION:<br><br>Provide the bleed path during feed and bleed cooling as credited in the PRA during loss of the secondary heat removal function. Scenario requires that injection for cooling be provided to the RCS and bled through the PORVs. The PORVs are located in the containment on the top of the pressurizer. There is one block valve for each PORV, which can be used to isolate the flowpath (for example, leaking PORV during normal operation). The pressure boundary function of the PORVs is safety related. The opening and closing of the PORVs during feed and bleed scenarios utilized the non-safety-related aspect of the PORVs (that is, disc travel, instrumentation and control). Control circuitry routing was identified in support of the fire PRA and, as such, is credited during fire scenarios.<br><br>FAILURE MODE: Fail to open on demand/fail to close on demand.<br><br>FAILURE RATE/PROBABILITY (FR/FP): 1E-01<br><br>FR/FP BASES: As discussed in the RC system notebook, operator action dominates this scenario (human error probability [HEP] assumed at 1E-01). |
| FUNCTIONS TO BE VERIFIED:<br><br>PORV opens on demand.<br><br>PORV (or block valve) closes on demand. |
| 10CFR50.69 TREATMENT PRACTICES:<br><br>As the feed and bleed function was identified as safety significant in the Maintenance Rule program, existing treatment practices provide reasonable confidence that the PORVs can reliably perform the function as credited in the categorization process. Thus, no additional treatment is required to support the 10CFR50.69 effort. |

### *B.2.2 Evaluation for Pressure Relief Function*

| SYSTEM: | FUNCTION: | MULTIPLE FUNCTIONS: |
|---|---|---|
| Reactor Coolant (RC) | Provide pressure relief during ATWS scenarios (active) | Yes |

SCENARIO DESCRIPTION:

The PRA credits the PORVs as opening during a number of anticipated transients without scram (ATWS) scenarios. They are also credited in the plant's risk assessment as part of the plant's response to the ATWS rule (10CFR50.62, *Requirements for the Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants*).

The PORVs along with the safety valves are credited in various combinations (for example, PORV only, safety valve only, etc.) to prevent overpressure of the RCS during ATWS events. The PORVs are located in the containment on the top of the pressurizer. There is one block valve for each PORV, which can be used to isolate the flowpath (for example, leaking PORV during normal operation). The pressure boundary function of the PORVs is safety related. The opening and closing of the PORVs during feed and bleed scenarios utilized the non-safety-related aspect of the PORVs (that is, disc travel, instrumentation and control). Control circuitry routing was identified in support of the fire PRA and, as such, is credited during fire scenarios.

FAILURE MODE: Fail to open on demand/fail to close on demand.

FAILURE RATE/PROBABILITY (FR/FP): 1E-01

FR/FP BASES: As discussed in the RC system notebook, operator action dominates this scenario (HEP assumed at 1E-01).

FUNCTIONS TO BE VERIFIED:

PORV opens on demand.

PORV (or block valve) closes on demand.

EXISTING TREATMENT PRACTICES:

The pressure relief function for ATWS scenarios was identified as safety significant during the development of the Maintenance Rule program (10CFR50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*).

10CFR50.69 TREATMENT PRACTICES:

As the pressure-relief function was identified as safety significant in the Maintenance Rule program, existing treatment practices provide reasonable confidence that the PORVs can reliably perform the function as credited in the categorization process. Thus, no additional treatment is required to support the 10CFR50.69 effort.

## B.3 Feedwater Control Logic System

This example illustrates the categorization and subsequent treatment for a water level control system installed in the feedwater system. Originally, this system was classified deterministically as non-safety related. After implementing 10CFR50.69, it was categorized as RISC-2. Further evaluation demonstrated that the existing treatment was still adequate.

| SYSTEM: | FUNCTION: | MULTIPLE FUNCTIONS: |
|---|---|---|
| Feedwater Control Logic | Control reactor pressure vessel water level (active) | Yes |
| SCENARIO DESCRIPTION:<br><br>As directed by the EOPs, provide for the control of the water level in the reactor pressure vessel (RPV) by controlling feedwater flow from the main condenser in conjunction with the condensate system. These scenarios typically include the loss of all other high-pressure injection sources (for example, high-pressure coolant injection [HPCI]) as well as the automatic depressurization (ADS) function.<br><br>FAILURE MODE: Fail to monitor level/fail to control level.<br><br>FAILURE RATE/PROBABILITY (FR/FP): 1E-02<br><br>FR/FP BASES: Plant-specific data from normal and off-normal startups and shutdowns | | |
| FUNCTIONS TO BE VERIFIED:<br><br>Monitoring of water level in the RPV.<br><br>Control (adjustment) of water level in the RPV. | | |
| EXISTING TREATMENT PRACTICES:<br><br>The non-safety-related feedwater level control system and the feedwater and condensate systems are used during each plant startup and most shutdown evolutions. This includes transitioning through all operating modes and maintaining water level during prolonged hot standby conditions where changes to water level may be quite frequent. This function is also included within the Maintenance Rule program as a safety-significant function. Additionally, the AP913 program lists these components as critical components (for example, due to their impact on plant generation/reliability) and has established associated performance and reliability measures (for example, preventive maintenance tasks). | | |
| 10CFR50.69 TREATMENT PRACTICES:<br><br>The existing treatment practices provide reasonable confidence that the feedwater control logic system can reliably perform the function as credited in the categorization process. | | |

## B.4 Fire Protection Pumps

This example is illustrates the categorization and subsequent treatment for fire protection pumps. Originally, this system was classified deterministically as non-safety related. After implementing 10CFR50.69, it was categorized as RISC-2. Further evaluation demonstrated that the existing treatment was still adequate.

<table>
<tr><td>SYSTEM:<br><br>Fire Protection</td><td>FUNCTION:<br><br>Flow and flow path to steam generators (active)</td><td>MULTIPLE FUNCTIONS:<br><br>Yes</td></tr>
<tr><td colspan="3">SCENARIO DESCRIPTION:<br><br>Provide flow to the steam generators through the main feedwater lines during severe accident conditions. The fire water pumps are located in a mild environment, and all equipment is outside containment. Spool pieces and associated tools are located at the fire-water-to-main-feedwater connection points.<br><br>FAILURE MODE: Fail to deliver flow.<br><br>FAILURE RATE/PROBABILITY (FR/FP): 1E-02<br><br>FR/FP BASES: Taken from the PRA notebook.</td></tr>
<tr><td colspan="3">FUNCTIONS TO BE VERIFIED:<br><br>Connect the fire water system to the main feedwater lines.<br><br>Fire water pumps deliver greater than 500 gpm (1892.5 liters per minute) flow.</td></tr>
<tr><td colspan="3">EXISTING TREATMENT PRACTICES:<br><br>The fire water system is normally a standby system servicing both nuclear safety-related and non nuclear-related functions. Due to property loss considerations, the system is maintained in accordance with National Fire Protection Association (NFPA) standards to ensure a high degree of reliability. This includes periodic testing and operation of the system. Additionally, the spool pieces and associated tools are contained in an operations department repetitive task sheet (RTS) and are verified once per fuel cycle.</td></tr>
<tr><td colspan="3">10CFR50.69 TREATMENT PRACTICES:<br><br>The existing treatment practices provide reasonable confidence that the fire protection system and associated operator actions can be performed as credited in the categorization process.</td></tr>
</table>

# C
# IMPACT ON PROCUREMENT PROCESSES

The purpose of this appendix is to describe the impact that adoption of 10CFR50.69 will have on procurement processes and to provide guidance on how to implement existing procurement processes for replacement items categorized under 10CFFR50.69.

## C.1 Nuclear Procurement Program Scope and Purpose

In general terms, the procurement processes at a nuclear power plant support operations and maintenance to ensure that the appropriate items are available when they are needed. The vast majority of procurements are for the following:

- Spare and replacement items for existing plant equipment
- Items needed for design modifications
- New plant equipment

The process is driven by the needs of maintenance or design engineering organizations. In turn, the needs are communicated to a technical group, often referred to as Procurement Engineering, who in concert with the Quality Assurance organization, specify the needed items. Additionally, these two organizations select an appropriate supplier, plan methods for how the item will be accepted for its intended use, and perform the acceptance activities.

## C.2 Basic Premises Regarding Procurement Processes

### *C.2.1 Suitability of the Design for Intended Applications*

10CFR50.69 does not change the design of an item through or as a result of the categorization process. As such, the Procurement Engineer need not question the adequacy of the original design and should use the original design requirements as a baseline from which to develop the technical procurement requirements for the replacement item unless these have been modified by a design modification. For example, if the original item was furnished with phenolic material, the Procurement Engineer should assume that phenolic is a suitable material for the intended application. If a replacement item is furnished with the identical phenolic material, then the Procurement Engineer should deduce that it is suitable for the application.

### *C.2.2 Use of Reputable Suppliers*

The guidance provided in this section assumes that the supplier that has been selected to furnish the item has been deemed a reputable supplier by the licensee. The degree of rigor necessary for making this determination is outside the scope of this document and should be consistent with existing utility practices for supplier evaluation and selection. Activities such as audits or commercial grade surveys are not required for approving or qualifying suppliers of items used in RISC-3 applications. Specification of supplier quality programs/processes may not be necessary and should be included in purchase documents at the discretion of each licensee.

### *C.2.3 RISC-3 Treatment vs. CGI Dedication*

The implementation of the procurement activities, which constitute alternative treatment for RISC-3 SSCs and their respective spare/replacement parts, will contribute to achieving reasonable confidence in the performance of design functions of those RISC-3 components. Although there are some similarities, this alternative treatment should not be construed as commercial grade item (CGI) dedication, which is a process controlled under each licensee’s 10CFR50 Appendix B Quality Assurance program. In other words, CGI dedication is not required for RISC-3 items.

### *C.2.4 Commercial Grade vs. Industrial Grade Material*

For the purposes of this report, the terms “commercial grade item” or “commercial grade material” may be used in the same context as the term “industrial grade material” (as denoted in the Statements of Consideration for 10CFR50.69).

## C.3 Overall Impact of RISC-3 Categorization on Procurement

As a result of the procurement initiatives begun in the 1990s, U.S. nuclear power plants have established procedures for the procurement of new and replacement items in support of plant operations and maintenance activities. In concert with this industry effort, EPRI has developed a series of guidelines to assist licensees with enhancing the efficiency of the procurement process and improving the overall quality of the items procured. The industry has realized significant improvements in ensuring that correct items are procured and accepted correctly the first time they are specified.

In general utility procurement processes focus on two primary aspects—the specification of the item and the acceptance of the item. Specification of the item is performed with a technical evaluation, which typically includes the classification of the item, evaluation of alternative items, and the actual specification of technical, quality, and supplier documentation requirements. The degree to which the technical evaluation may be treated differently for items categorized under 10CFR50.69 is discussed in Section C.4. For the most part, the same process should be used to evaluate alternative (non-identical) replacement items and to develop their specifications. The level of detail and/or the amount of technical information communicated to the supplier in the purchase document will vary among RISC-3 items and may be less than that typically specified for other safety-related items.

The degree to which the acceptance process may be performed differently for items categorized under 10CFR50.69 is discussed in Section C.5. Because this process is no longer subject to the requirements of 10CFR50 Appendix B for RISC-3 items, there are significant areas where the process may be modified without decreasing the confidence that the licensee has regarding the item’s ability to perform designated functions in RISC-3 applications.

## C.4 Performing the Technical Evaluation

The process used to correctly specify an item is referred to as a technical evaluation. Each licensee’s existing technical evaluation process should be used to evaluate and specify items categorized under 10CFR50.69. Furthermore, the categorization process is assumed to have been completed.

The user of this report should note that the technical evaluation process is used by many licensees for specifying and evaluating both safety-related and non-safety-related items, with less stringent reviews and lower levels of documentation being used for non-safety-related items.

### *C.4.1 Component Categorization and Subcomponent Classification*

#### C.4.1.1 Component Categorization

For the purposes of this report, the process for categorizing an SSC under 10CFR50.69 is not discussed further. The categorization process is assumed to have been completed. From a procurement perspective, Procurement Engineers should assume that the resulting categorization is appropriate and should use the technical evaluation process described in this report.

#### C.4.1.2 Subcomponent Classification

The adoption of 10CFR50.69 does not require the licensee to use PRA techniques for the classification or reclassification of spare/replacement parts. The deterministic techniques, as employed by many licensees through the use of the failure modes and effects analysis, remain a viable process for classifying items at the subcomponent level. Currently, most licensees use a system for identifying the procurement category for spare/replacement parts as illustrated in Figure C-1, based on the part either being classified as safety related or non-safety related.

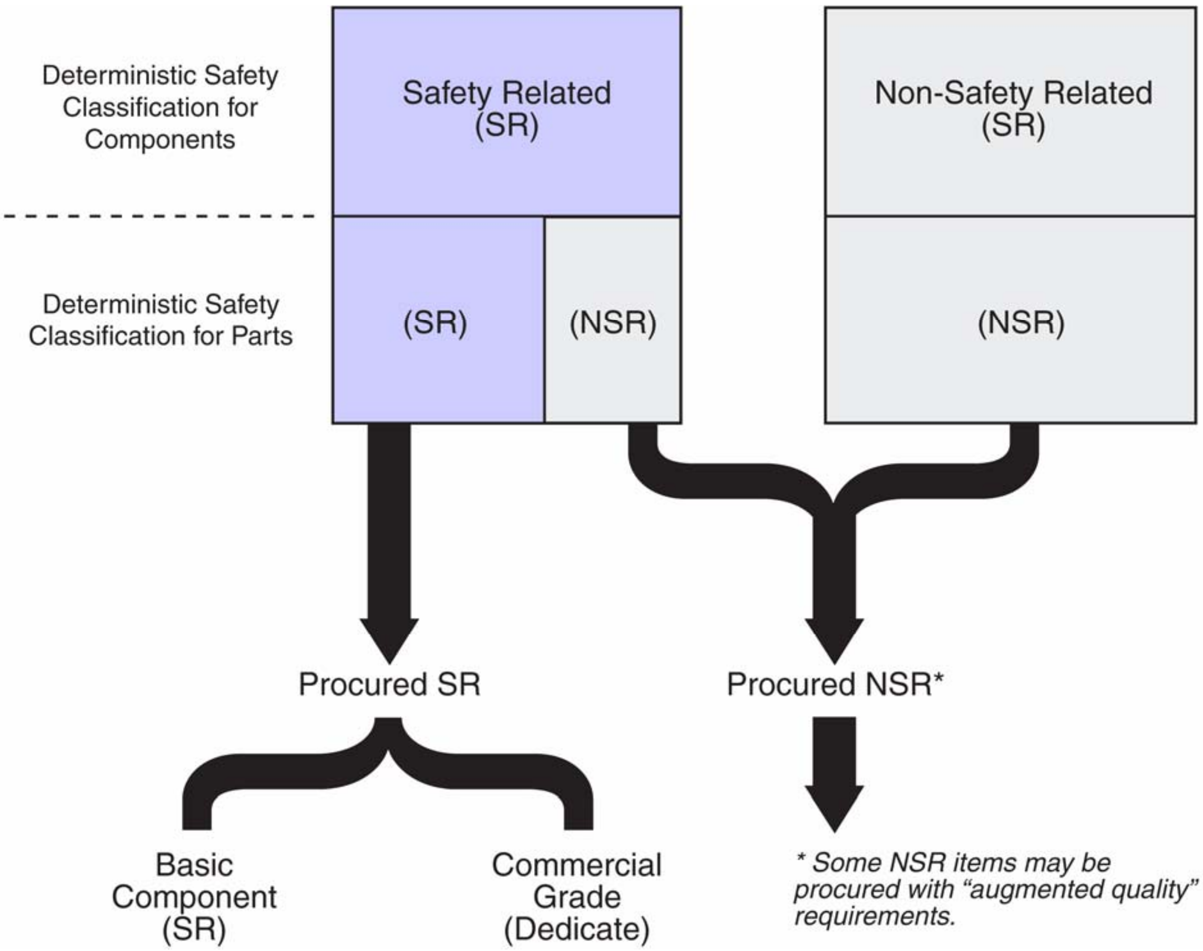


**Figure C-1**
**Current Procurement Categories for Most Licensees**

Figure C-1 illustrates that for safety-related components, subcomponents may be classified as either safety related or non-safety related, based upon their function within the component. The figure also shows that safety-related subcomponents may be procured either as a basic component (that is, from an organization with a 10CFR50 Appendix B QA program) or commercial grade and dedicated by the licensee.

Figure C-2 illustrates how the classification of subcomponents changes under 10CFR50.69. For spare/replacement items intended for use in RISC-3 components, the licensee may classify these items as non-safety related, given that the alternative treatments described in this report are implemented and applied. The subcomponents may be considered as non-safety related because 10CFR50 Appendix B no longer applies to the host component. This change affords the licensee an opportunity to procure more items as non-safety related and fewer items as safety related.

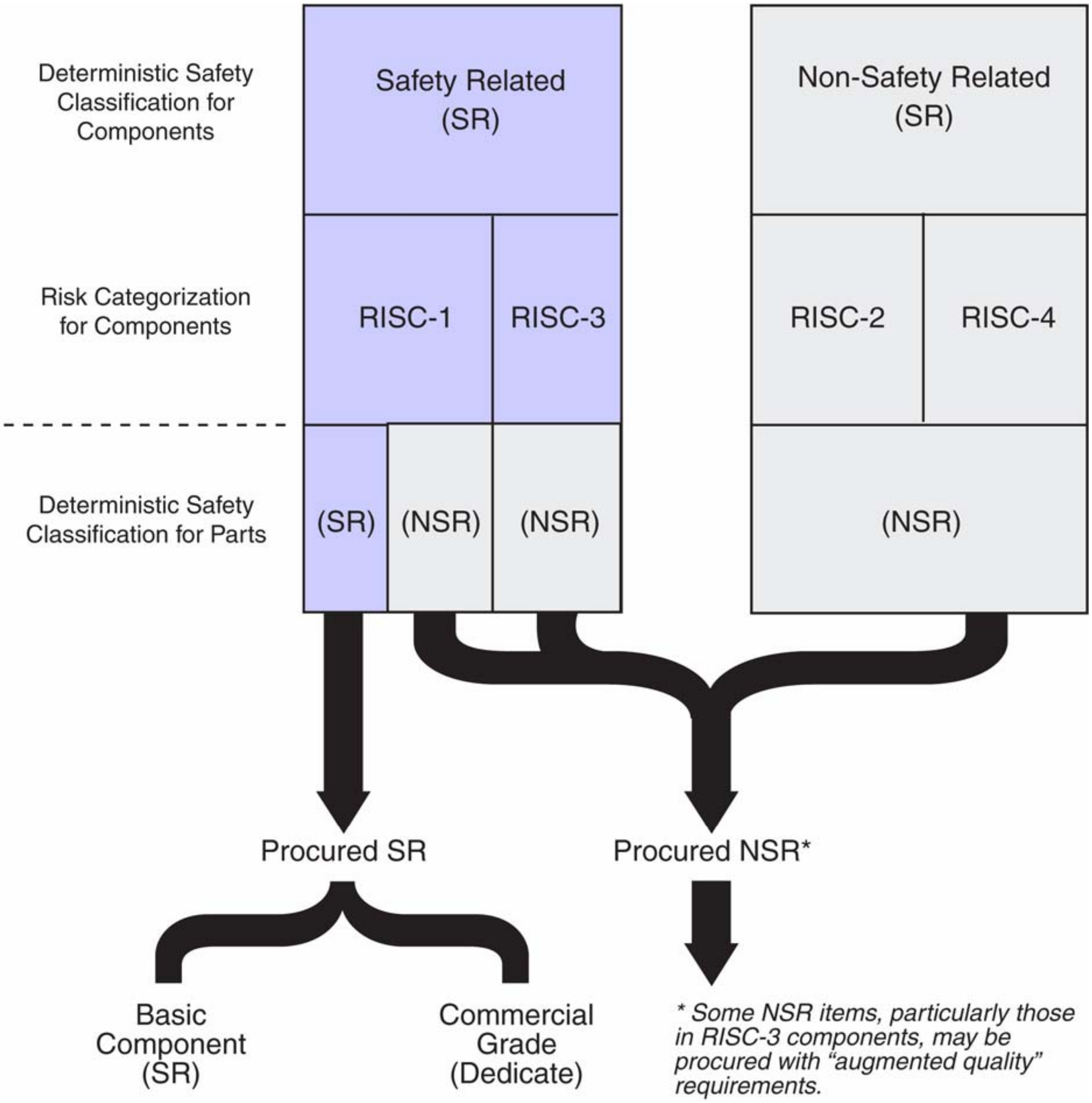


**Figure C-2**
**New Subcomponent Procurement Categories under 10CFR50.69**

For spare/replacement items intended for use in RISC-1 components, the licensee should ensure that the technical justification for classifying (or "downgrading") certain subcomponents is still valid, given the risk-informed safety significance of the host component. In most cases, the classification of parts in RISC-1 components will not change. For spare/replacement items intended for use in RISC-2 or RISC-4 components, the licensee may continue to procure these items as non-safety related.

The licensee has the option for any non-safety-related subcomponent to treat the procurement as "augmented quality," which allows the licensee to select appropriate quality controls based on the function and importance of the item and its host component. The use of augmented quality practices would be most appropriate when procuring non-safety-related subcomponents intended for use in RISC-3 components.

### C.4.2 Determining the Procurement Scenario

Procurements can be grouped into one of the following three scenarios:

- Procurement of an identical replacement item (like-for-like procurement)
- Procurement of an equivalent replacement item (alternative item procurement)
- Procurement of a non-equivalent item or a new item that has been deemed suitable for its intended application through plant design activities

The process for determining the appropriate procurement scenario is performed on a case-by-case basis and already exists at each plant. Categorization and implementation of 10CFR50.69 does not affect how this process should be implemented.

### C.4.3 Technical Evaluation for Identical Items

Categorization and implementation of 10CFR50.69 does not affect how this process should be implemented. Identical items should be specified using the same technical and quality procurements as the last time the item was procured.

### C.4.4 Technical Equivalency Evaluation for Alternative Items

Existing EPRI guidance and many licensee procedures describe two approaches for evaluating an alternative item:

- Determining whether it is equivalent to the original
- Determining whether it is suitable for its intended application(s)

Determining the most appropriate approach should be based upon the following factors:

- The type and extent of physical changes made to the alternative item
- The complexity of the item
- The accident function(s) of the replacement item, if applicable

Similar to other processes associated with the technical evaluation, 10CFR50.69 does not affect how these processes are implemented. As such, the licensee should continue to use their existing process for evaluating alternative items, regardless of whether they have adopted 10CFR50.69 and categorized SSCs accordingly.

Additional guidance regarding the evaluation of alternative replacement items intended for use in components designed to withstand a seismic event is provided in the following reports:

- EPRI Report NP-7484, *Guideline for the Seismic Technical Evaluation of Replacement Items for Nuclear Power Plants*
- EPRI Report TR-105849, *Generic Seismic Technical Evaluations of Replacement Items for Nuclear Power Plants--Item-Specific Evaluations*
- EPRI Report 1011783, *RISC-3 Seismic Assessment Guidelines*

Additional guidance regarding the evaluation of alternative replacement items intended for use in components designed to withstand harsh environments is provided in EPRI Report 1008748, *Guidance for Accident Function Assessment for RISC-3 Applications*.

Additional guidance regarding the identification of critical characteristics for design is provided in EPRI Report TR-102260, *Supplemental Guidance for the Application of EPRI Report NP-5652 on the Utilization of Commercial Grade Items*.

### C.4.5 Technical Evaluation for New Items

Categorization and implementation of 10CFR50.69 does not affect how this process should be implemented. New items (or non-equivalent replacement items deemed suitable for their intended application via the design process) should be specified by translating the requirements resulting from the design activities.

### C.4.5 Specification Guidance

The final step leading to the actual specification of an item intended for use in a component categorized under 10CFR50.69 is the selection of the supplier. In many cases, the supplier will be one that has furnished items in the past. Upon completion of the simplified technical evaluation, sufficient information will have been generated for the Procurement Engineer to prepare the purchase requisition. Like purchase requisitions for most safety-related items, the requisition should include the following procurement requirements that are communicated to the supplier in the purchase order:

- Technical requirements
- Supplier quality program requirements
- Supplier documentation requirements

#### C.4.5.1 Specifying Technical Requirements

The technical requirements specified are not affected by the categorization of the item according to 10CFR50.69. As such, the technical requirements specified in the purchase order should not change because of the categorization.

### C.4.5.2 Specifying Supplier Quality Program Requirements

Items categorized as RISC-1 should be procured with the same quality requirements as before the adoption of 10CFR50.69. Items categorized as RISC-2, though non-safety related, should be procured with the same quality requirements, if any, as before the adoption of 10CFR50.69.

RISC-3 activities are not required to be performed under a 10CFR50 Appendix B Quality Assurance program, and suppliers of RISC-3 items are not required to have an audited and approved quality assurance program. However, certain activities on the part of the supplier/manufacturer may be desirable. Some examples may include the following:

- A written verification may be requested that specific materials have been used or excluded from use in manufacture.
- If there is a specific attribute of the component being purchased that is unusual or not commonly purchased, the licensee can still request the vendor to confirm that the attribute has been considered in the design or manufacture of the component.
- The licensee may specify a particular vendor procedure describing a special process (for example, welding, coating, heat treatment).

Procurement documents should not specify “nuclear safety related” or be controlled by a 10CFR50 Appendix B Quality Assurance program. Most suppliers have a commercial Quality Assurance program. Including requirements for application of the commercial Quality Assurance program in the specification is at the option of the licensee.

### C.4.5.3 Specifying Required Supplier Documentation

The documentation required to furnish objective evidence that the technical requirements of items and the supplier’s quality requirements have been met should be specified. The amount of documentation sufficient to provide reasonable confidence will vary, however, depending on the component categorization resulting from 19CFR50.69. As a general rule of thumb, supplier documentation is most appropriate for RISC-1 items, but it is less important for items intended for use in the other categories of equipment.

The supplier may be requested to affirm that the technical requirements of the purchase order have been met. Suppliers have various means of providing affirmation, which may include an implicit acceptance of the purchase order or a letter. Most suppliers currently provide a certificate of conformance in response to such requests.

## C.5 Acceptance of Items After Adoption of 10CFR50.69

### *C.5.1 Acceptance of Items Intended for Use in RISC-1 Components*

Items intended for use in components categorized as RISC-1 should be accepted using the same special treatment as before the adoption of 10CFR50.69. This special treatment will either involve the acceptance of items procured directly from a 10CFR50 Appendix B supplier or dedication of items procured as commercial grade. The acceptance processes should not change as a result of adoption of 10CFR50.69 for RISC-1 items.

### *C.5.2 Acceptance of Items Intended for Use in RISC-2 Components*

Items intended for use in components categorized as RISC-2, though non-safety related, should be accepted using the same treatment, if any, as before the adoption of 10CFR50.69.

### *C.5.3 Acceptance of Items Intended for Use in RISC-3 Components*

Because 10CFR50 Appendix B and 10CFR21 do not apply to RISC-3 SSCs or their spare/replacement items, the associated requirements for the acceptance of basic components and commercial grade items used in safety-related applications do not apply. The licensee should ensure that that appropriate actions are taken to achieve reasonable confidence that the procured item intended for use in RISC-3 components meets specified requirements.

Depending on the complexity and sophistication of the RISC-3 component being procured, receipt inspections/tests, crediting existing supplier evaluations, or reliance on the historical performance of the item and the supplier may be used to achieve reasonable confidence that the device is acceptable and will perform its accident function. All domestic utilities have processes for accepting items intended for both safety-related and non-safety-related applications. As such, each licensee should assess the cost effectiveness and benefits of using similar or simplified processes as a means for accepting these items. Table C-1 illustrates alternative treatment to achieve a level of confidence appropriate for items intended for use in RISC-3 components. In any option, or combination of options selected by the licensee, the level of documentation should also be adjusted as deemed necessary based on the low safety significance of the procured item.

**Table C-1**
**Acceptance Bases for Items Intended for Use in RISC-3 Components**

| Acceptance Method | Recommended Acceptance Process Modifications for RISC-3 Items |
|---|---|
| Receipt inspections/tests of procured items | • Use reduced sampling plans |
| OR | |
| Supplier commercial quality activities | • Reference existing supplier evaluations (that is, supplier audit reports, commercial grade surveys, etc.) that are available to the licensee.<br><br>Note: The licensee will have experience with the supplier and will know whether they are reputable and reliable. There are no mandatory supplier qualification requirements associated with RISC-3 procurements. As with any procurement, licensees should be comfortable with a new supplier's capabilities before purchasing an item. However, suppliers of RISC-3 items do not have to be audited or maintain an approved 10CFR50 Appendix B quality assurance program. Likewise, audits and/or commercial grade surveys are not required for approving or qualifying new suppliers of items used in RISC-3 applications. |
| OR | |
| Reliance on the historical performance of the item and the supplier | • Good supplier performance history and/or item performance history may be credited for establishing reasonable confidence |

### C.5.4 Acceptance of Items Intended for Use in RISC-4 Components

Items intended for use in components categorized as RISC-4 would typically not have had any acceptance requirements because they were originally classified as non-safety related and are now categorized as low safety significant from a risk-informed perspective. As such, there is no need to enhance existing acceptance, and in most cases, acceptance/verification activities are not necessary.

## C.6 Reporting of Defects per 10CFR21 for Licensees

### C.6.1 Overview of Relevance of 10CFR21 to the Risk-Informed Approach

The rule provides the following overview regarding the development and relevance of 10CFR21:

> Section 50.69 substitutes a risk- informed approach for regulating nuclear power plant SSCs for the current deterministic approach. Therefore, it is necessary from the standpoint of regulatory coherence to determine:
>
> (1) what categories of SSCs (i.e., RISC-1, RISC-2, RISC-3, and RISC-4) should be subject to part 21 and § 50.55(e) reporting under § 50.69 and whether changes to part 21 and/or § 50.55(e) are necessary to ensure proper reporting of substantial safety hazards caused by these SSCs; and
>
> (2) the appropriate reporting obligations of licensees and vendors under § 50.69, and whether changes to part 21 and/or § 50.55(e) are necessary to impose the intended reporting obligations on these entities under § 50.69.

### C.6.2 Applicability of 10CFR21 to Risk-Informed Categorized SSCs

Table C-2 summarizes the applicability of 10CFR21 to each category of SSC described in the rule.

**Table C-2**
**10CFR21 and Risk-Categorized SSCs**

| SSC Category | Does 10CFR21 Apply? |
|---|---|
| RISC-1 | Yes |
| RISC-2 | No |
| RISC-3 | No |
| RISC-4 | No |

The applicability of 10CFR21 is the same for nuclear suppliers as it is for licensees.

## C.7 Procurement Examples

The examples in this section are provided for illustrative purposes only and describe actions that would provide reasonable confidence that a replacement item was procured appropriately. The amount of technical detail specified for a given item and the rigor involved in the acceptance of that item should be determined on a case-by-case basis and should comply with plant-specific procedures and guidance.

### *C.7.1 Procuring an Identical Replacement Component*

In this example, the utility is required to replace a 300# (PN 50) globe valve. The valve was originally classified as safety related and has since been categorized using the NEI 00-04 process as RISC-3. If an identical valve is available from the original equipment manufacturer, it can be used provided the following are met:

- The valve is specified with the same technical requirements as when it was originally procured.
- The supplier’s commercial quality program/quality controls are invoked in the procurement document.
- The valve is accepted at receipt with a QC receipt inspection to ensure that the appropriate documentation was furnished and that the item meets the requirements specified in the procurement document.

### *C.7.2 Procuring an Equivalent Replacement Item*

In this example, the utility is required to procure an O-ring in a torque switch. The torque switch was originally classified as safety related and has since been categorized using the NEI 00-04 process as RISC-3. An identical O-ring is no longer available from the original equipment manufacturer, but an O-ring of an alternative material is offered. The alternative O-ring may be used provided:

- An item equivalency evaluation is performed to determine whether the alternative item is equivalent to the original O-ring. (For the purposes of this example, the alternative O-ring material was determined to be suitable.)
- The O-ring is specified with technical requirements reflecting its current design (that is, same dimensions, but different material, part number, etc.)
- The supplier’s commercial quality program/quality controls are invoked in the procurement document
- The O-ring is accepted at receipt with a QC receipt inspection to ensure that the appropriate documentation was furnished and that the item meets the requirements specified in the procurement document

### *C.7.3 Procuring a New RISC-3 Component*

In this example, the utility is required to procure a new lube oil pump that has been categorized using the NEI 00-04 process as RISC-3:

- The new lube oil pump is specified with technical requirements reflecting the design deemed suitable (by the design engineering organization) for its intended application.
- The supplier's commercial quality program/quality controls are invoked in the procurement document.
- The lube oil pump is accepted at receipt with a QC receipt inspection to ensure that the appropriate documentation was furnished and that the item meets the requirements specified in the procurement document.

### *C.7.4 Procurement Example of Cost Benefit*

Table C-3 illustrates a number of examples demonstrating the potential savings that may be realized by categorizing and treating a safety-related item as RISC-3 in lieu of procurement options currently available for a safety-related item. The costs shown in the table represent total procurement costs (purchase price plus acceptance actions). These examples are provided for illustrative purposes only and should be used as a means to compare relative costs. Actual costs will vary somewhat from those shown due to numerous factors associated with each licensee's procurement practices.

**Table C-3**
**10CFR21 and SSCs Categorized from a Risk-Informed Perspective**
**(Courtesy of Dominion Generation)**

| | **Deterministic Classification** | | **Risk-Informed Categorization** |
|---|---|---|---|
| | **Safety-Related** | | **RISC-3** |
| **Procured Items** | **Procured as a Basic Component** | **Procured as Commercial Grade and Dedicated** | **Procured as Commercial Grade and Treated per EPRI 1001234** |
| Relief Valve, 1 ½" X2" | $11,000 | $4,400 | $3,600 |
| Operator (Valve) | $30,000 | $15,000 | $9,900 |
| Gate Valve, 3" SS | $7,000 | $800 | $130 |
| Butterfly Valve, 36" | $36,000 | $13,000 | $9,500 |
| Operator (Large Bore Valve) | $70,000 | $23,000 | $18,000 |
| Check Valve | $3,200 | $1,000 | $320 |

# D
# APPLICABILITY OF GUIDANCE CONTAINED IN GENERIC LETTER 91-18 REGARDING OPERABILITY

The purpose of this appendix is to describe the impact that adoption of 10CFR50.69 may have on the operability of SSCs categorized and treated under 10CFR50.69.

## D.1 Introduction

Generic Letter 91-18, "Degraded and Non-Conforming Conditions and Operability," forwarded two sections of the NRC Inspection Manual on degraded and non-conforming conditions and operability to licensees. The sections were forwarded "to ensure consistency in application of this guidance by the NRC" because there had been differences in application in the past by NRC staff. The following text is not meant to be an exhaustive discussion of GL91-18 with respect to RISC-3 components, but rather to provide a general summary of its applicability.

## D.2 Inspection Manual Section: "Resolution of Degraded and Nonconforming Conditions"

This inspection manual section defines a degraded condition as "A condition of an SCC in which there has been any loss of quality or functional capability." A nonconforming condition is defined as follows:

> A condition of an SSC in which there is failure to meet requirements or licensee commitments

Full qualification is defined as:

> . . . conforming to all aspects of the current licensing basis, including codes and standards, design criteria and commitments.

The definition of full qualification contains a key phrase: "conforming to all aspects of the current licensing basis." When plant adopts 10CFR50.69 and SSCs undergo risk-informed categorization and treatment, the current licensing basis changes. For RISC-3 components, the special requirements for environmental qualification under 10CFR50.49 and quality assurance under 10CFR50 Appendix B no longer apply. For RISC-3 components, accident function assessment occurs under design and procurement activities, which need not be controlled by 10CFR50.49 or by industry standards applicable to safety-related equipment.

The definition of nonconforming condition has four examples. Their applicability is shown in Table D-1.

**Table D-1**
**10CFR21 and SSCs Categorized from a Risk-informed Perspective**

| Example | Example of Nonconformance | Applicability |
|---|---|---|
| 1 | There is failure to conform to one or more applicable codes or standards specified in the FSAR. | There should be no codes or standards in the FSAR that are applicable to RISC-3 components |
| 2 | As-built equipment, or as-modified equipment, does not meet FSAR design requirements. | The FSAR may no longer need to specify design requirements for RISC-3 components. (Each licensee should take care to review plant-specific commitments relative to FSAR content and revisions.) |
| 3 | Operating experience or engineering reviews demonstrate a design inadequacy. | If an RISC-3 component is found to be degraded in service to a point that<br>1) Its ability to function under accident conditions is questionable or<br>2) A design inadequacy is discovered that makes the ability to function under accident conditions questionable.<br>Then the operability of the component would be compromised and actions required under the license (for example, LCO requirements) would be applicable. |
| 4 | Documentation required by NRC requirements such as 10CFR50.49 is not available or is deficient. | 10CFR50.49 does not apply to RISC-3 components. Accident function assessment applies; however, there is no requirement to formally document accident function assessment. |

With regard to the examples of nonconforming conditions from the inspection manual, the identification of a significant degraded condition or inadequacy in the accident function assessment constitutes a concern for operability with respect to the Technical Specification requirements. Any limiting condition for operation (LCO) applicable to a RISC-3 component should be addressed, and a determination should be made as to whether the degraded device needed to be replaced or modified and/or the accident function assessment corrected or upgraded. RISC-3 applications with the same conditions would most likely be modified accordingly.

With regard to identifying an inadequacy in the accident function assessment, it is possible that a deficiency in the basis of an accident function assessment for a component could be identified in service. Should such a condition be identified, a correction would be required, or an alternative replacement device with an adequate basis would have to be installed. Again, other RISC-3 applications with the same conditions would be modified accordingly.

Both Section 4.5.3, “Items for Consideration in a JCO,” and Section 4.6, “Reasonable Assurance of Safety,” of the inspection manual section indicate that:

1. “Probability of needing the safety function,” and
2. “PRA or Individual Plant Evaluation (IPE) results that determine how operating the facility in the manner proposed in the JCO will impact the core damage frequency,”

should be considered in the assessment of continuing plant operation. By definition, RISC-3 components have been determined to have low risk significance.

## D.3 Inspection Manual Section: “Operable/Operability: Ensuring the Functional Capability of a System or Component”

This section uses the same definitions of degraded condition, nonconforming condition, and full qualification.

Section 4.0, “Background,” states in part:

> Without any information to the contrary, once a component or system is established as operable, it is reasonable to assume that the component or system should continue to remain operable, and the previously stated verifications should provide that assurance. However, whenever the ability of a system or structure to perform its specified function is called into question, operability must be determined from a detailed examination of the deficiency.

In other words, for RISC-3, operability is a concern when the ability to perform the accident function is in question.

Section 5.0, “Additional Guidance for Operability Determinations,” provides three conditions for invoking an operability determination. These are examined in Table D-2.

**Table D-2**
**Applicability of Operability Determination Requirements for RISC-3 Components**

| Example | Operability Determination Requirement | Applicability |
|---|---|---|
| 1 | Discovery of degraded conditions where performance is called into question | Per previous discussion, this item applies to RISC-3 components. |
| 2 | Discovery of nonconforming conditions where the qualification of equipment (such as conformance to codes and standards) is called into question | Conformance to codes and standards does not apply. However, if the accident assessment were found to be significantly incorrect or inadequate, the assessment would have to be upgraded and operability assessed. |
| 3 | Discovery of an existing but previously unanalyzed condition or accident | Operability and RISC-3 status would have to be assessed and the licensee needs to ensure that current rule issues are considered. |

The concept of full qualification changes with the application of 10CFR50.69, and the concern for conforming to codes and standards is eliminated. However, design criteria and any commitments for components stated in the FSAR remain in effect. Therefore, if design criteria and/or commitments are not met in an accident functional assessment, an operability issue exists. This section of the inspection manual refers the reader back to the JCO section of “Resolution of Degraded and Nonconforming Conditions” described above.

Section 6.2, “Treatment of Single Failures in Operability Determinations,” would apply to RISC-3 components that have been determined to be degraded or nonconforming. Accordingly, components of the same type that are subject to the same or more severe conditions would have to be reviewed with respect to the degradation or nonconformance.

Section 6.9, “Use of Probabilistic Risk Assessment in Operability Decisions,” states that a PRA is not acceptable for making operability decisions. The section then states: “However, PRA may provide valid and useful supportive information for a license amendment. The PRA is also useful for determining the safety significance of SCCs.”

Section 6.10, “Environmental Qualification,” states that the licensee must make a prompt determination of operability if a potential deficiency in the environmental qualification of the equipment is identified. As stated above, identification of significant degradation of the installed component or identification of an inadequacy in the accident function assessment would require an operability assessment.

A note at the end of Section 6.10 indicates that reporting under 10CFR50.72, 10CFR50.73, 10CFR21, and 10CFR50.9 should be evaluated for applicability. 10CFR50.69(1) explicitly states that 10CFR21, 10CRF50.72, and 10CFR50.73 do not apply to RISC-3 components and that no reporting under these sections is required. 10CFR50.9 states that the “licensee shall notify the Commission of information identified by the licensee as having for the regulated activity a significant implication for the public health and safety or common defense and security." Given that RISC-3 components have low risk significance, any problems associated with them would have little implication for adverse effects on "the public health and safety or common defense and security."

## D.4 Summary

The licensee should consider operability issues for those RISC-3 components where significant degradation of the in-service component or significant inadequacy in the accident function assessment has been identified. As noted in preceding sections of this report, the reporting requirements of 10CFR21, 10CFR50.9, 10CRF50.72, and 10CFR50.73 are not applicable to RISC-3 components.

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