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June 15, 2001

Mr. David B. Matthews
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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Matthews:

Enclosed for NRC staff review is Draft Revision B to NEI 00-04, *Option 2 Implementation Guideline.* This draft addresses the NRC observations and comments provided in your April 5, 2001, letter.

Enclosure 1 is a line-in/line-out version of the guideline that indicates all changes made from the previous draft. Enclosure 2 is a clean version. Enclosure 3 is a table listing the NRC comments and a summary of the NEI response.

We look forward to discussing the revised guideline and the pilot project at our scheduled June 27 meeting. If you or your staff have any immediate questions, please contact Adrian Heymer (202)-739-8094, e-mail aph@nei.org or me.

Sincerely

Anthony R. Pietrangelo

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Enclosures

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NEI 00-04 (DRAFT - Revision B)

Option 2 Implementation Guideline

NUCLEAR ENERGY INSTITUTE

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1 BACKGROUND

The regulations for design and operation of US nuclear plants define a specific set of accidents that the plants must tolerate without incurring significant public health impacts. This is known as a deterministic regulatory basis because there is ~~no~~ little explicit consideration of the probability of occurrence of the design basis accidents – it is "determined" they will occur, and the plant is designed and operated to prevent and mitigate such accidents. This deterministic regulatory basis was developed over thirty years ago, absent data from actual plant operation. It is based on the principal that the deterministic accidents would serve as a surrogate for the broad set of transients and accidents that could be realistically expected over the life of the plant.

Since the inception of this regulatory basis, over 2500 reactor years of operation have been accumulated in the US (over 9000 reactor years worldwide), with a corresponding body of data relative to actual transients, accidents, and plant equipment performance. Further, each US plant has performed a probabilistic risk analysis (PRA), which uses these data, and models a large number of potential accident sequences (including sequences not considered in the deterministic regulatory basis) to estimate the overall risk from plant operation. PRAs describe risk in terms of the frequency of reactor core damage and/or significant offsite release. Insights from PRAs reveal that certain plant equipment important to the deterministic regulatory basis is of little significance to safety. Conversely, certain plant equipment is important to safety but is not included in the deterministic regulatory basis.

Risk insights have been considered in the promulgation of new regulatory requirements (e.g., station blackout rule, anticipated transients without scram rule, maintenance rule). In 1998, the Commission directed the NRC staff to develop rulemaking to more broadly consider risk insights as the basis for fundamental reform to the deterministic regulatory approach. This guideline addresses the use of risk insights to define the scope of plant equipment subject to special regulatory treatment provisions.

1.1 REGULATORY REFORM INITIATIVE

Current NRC regulations establish that plant equipment necessary to meet the deterministic regulatory basis is categorized as "~~safety related~~safety-related", and is subject to a broad set of "special treatment[1]" regulations (controls). Other plant equipment is categorized as "non ~~safety related~~safety-related", and is not subject to special regulatory treatment. There is a set of non~~safety related~~safety-related equipment that is subject to the regulations and a degree of special treatment. This set is often referred to as "important-to-safety."

[1] Special treatment requirements are current requirements imposed on structures, systems, and components that go beyond industry-established requirements for equipment classified as commercial grade that are intended by the NRC to provide additional confidence that the equipment is capable of meeting its functional requirements under design basis conditions. These additional special treatment requirements include additional design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements.

The objective of regulatory reform is to adjust ~~the scope~~the scope of equipment subject to special regulatory treatment (controls) in light of risk insights from PRAs and plant operation. This will result in adjustment of controls based on the safety-significance of the equipment.

When issued, 10 CFR 50.69, will ~~provide an~~provide an option for licensees to implement a risk-informed approach for regulations that establish special treatment requirements for plant structures, systems and components (SSCs) based on safety significance. Table 1.1 lists the special treatment regulations that would be subject to the optional risk-informed approach. 10 CFR ~~50.69 defines~~50.69 defines four categories of SSCs, based on existing safety classification and risk significance, and establishes ~~controls as~~controls as a function of the categorization. The special treatment regulations in Table 1.1 would not in themselves be changed. However, the scope of applicability, and the manner in which the special treatment provisions are implemented, would be revised as defined in 10 CFR 50.69.

The decision to adopt a risk-informed approach for categorizing structures, systems and components is voluntary. Each licensee will make its determination on whether to adopt a risk-informed approach to regulation based on the estimated benefit. 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 scheme should result in an increased awareness on that set of equipment and activities that could impact safety.

The NRC rulemaking plan does not replace the existing "~~safety related~~safety-related" and "non ~~safety related~~safety-related" classifications. Rather, 10 CFR 50.69 provides that the each existing classification category can be divided into two categories based on high or low ~~safety significance.~~safety significance. The categorization is depicted in Figure 1.1-1.

The application of special treatment regulations and controls is a function of the categorization. Regulatory treatment requirements are not applicable ~~applied for all categories except~~ to RISC-4 SSCs. The existing special treatment provisions for RISC-1 and RISC-2 are maintained or enhanced. RISC-3 equipment would be subject to the licensee's standard commercial (balance-of-plant) controls with monitoring to provide reasonable assurance that the function directly referenced in the regulations or in the safety analyses required by regulation will be satisfied.

Figure 1.1-1
Risk Informed Safety Classifications (RISC)

High	**RISC-1 SSCs** Safety Related, Safety Significant *Reliability Assurance*	**RISC-2 SSCs** Non-Safety Related, Safety Significant *Reliability Assurance*
Low	**RISC-3 SSCs** Safety Related, Low Safety Significant *Maintain Function* *Commercial (BOP) Programs*	**RISC-4 SSCs** Non-Safety Related, Low Safety Significant *Commercial (BOP) Programs*
	Safety Related	Non-Safety Related

DETERMINISTIC SIGNIFICANCE

1.2 CATEGORIZATION PATHWAYS

The risk-informed classification scheme ~~allocates each~~ assigns SSCs ~~in the plant~~ to one of four classifications (RISC-1 – 4). Figure 1.2-1 provides a graphical depiction of the classification pathways utilized in this process. The existing ~~safety related~~safety-related components in the plant are classified either via Pathway A to RISC-1 (for safety significant SSCs) or via Pathway B to RISC-3 (for low safety significant SSCs). Pathway A is the default pathway for all ~~safety related~~safety-related SSCs. That is, unless a compelling case can be made that the ~~safety related~~safety-related SSC is low safety significant, then it is classified as RISC-1. In cases where a risk-informed process can demonstrate that the ~~safety related~~safety-related SSC is of low safety significance, it is classified as RISC-3.

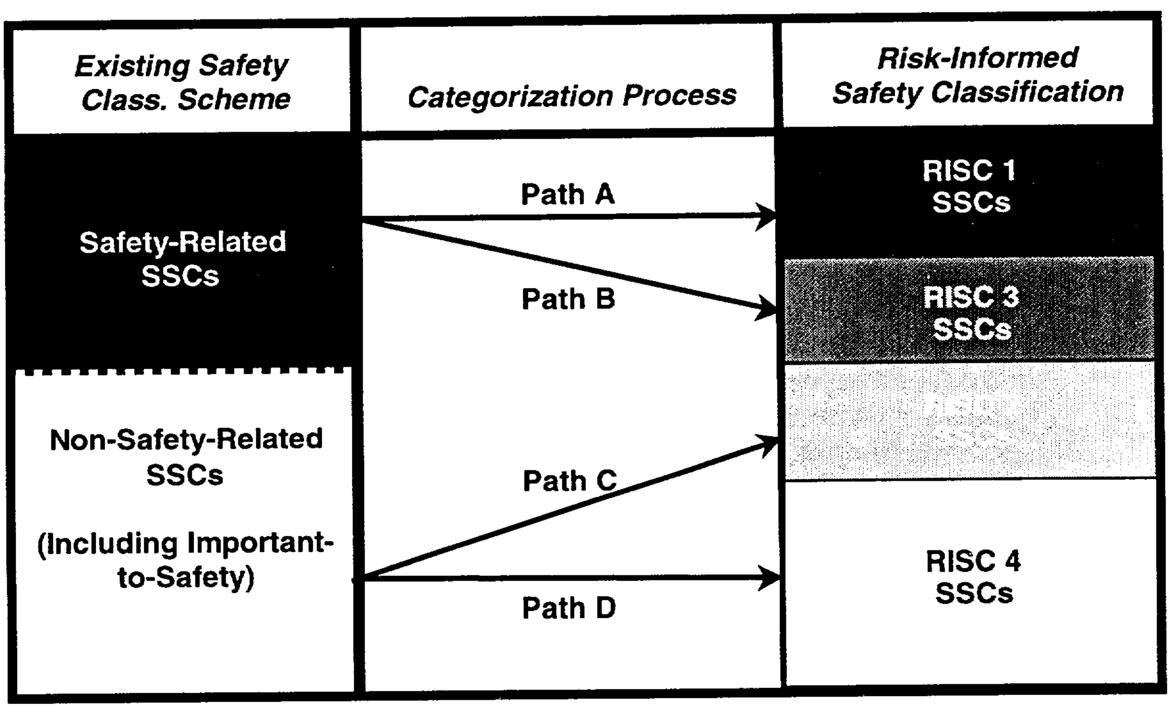
All other SSCs (non-~~safety related~~safety-related) are classified on either Pathway C to RISC-2 (for safety significant SSCs) or via Pathway D to RISC-4 (for low safety significant SSCs). In this case, Pathway D is the default pathway for non-~~safety related~~safety-related SSCs. That is, unless a compelling case can be made that the non-~~safety related~~safety-related SSC is safety significant, then it is classified as RISC-4. In cases where a risk-informed process can demonstrate that the non-~~safety related~~safety-related SSC is safety significant, it is classified as RISC-2.

The only time that an SSC would move via another pathway would be if it was found to be misclassified initially. In that case, the licensee would use its configuration control (design control element) including the application of §~~perform a~~ §50.59.~~-evaluation to reclassify the SSC as nonsafety related.~~

Certain plant equipment is not considered ~~safety related~~safety-related in the existing classification scheme, but is subject to certain special regulatory requirements. Examples are, "important to safety" SSCs, whose failure could affect the function of ~~safety related~~safety-related SSCs, or "augmented quality" SSCs that require some subset of "~~safety related~~safety-related" regulatory treatment (e.g., many plants consider fire protection SSCs as augmented quality).

For the purposes of regulatory reform, these "important to safety" SSCs as described above enter into the categorization process as "non ~~safety related~~safety-related". However, their default pathway is not into RISC-4. Rather, the default pathway is into RISC-2, with the assumption that the existing regulatory requirements would be maintained, absent compelling justification to change them. Thus, requirements for these "important-to-safety" SSCs would not change. If the risk informed classification process determines that these SSCs have low safety significance, they may be classified to RISC-4.

Figure 1.2-1
CONCEPTUAL REPRESENTATION OF CLASSIFICATION PATHWAYS



1.3 IMPLEMENTATION PROCESS

This document provides detailed implementation guidance for 10 CFR 50.69.~~:~~

If the licensee determines that its implementation process satisfies §50.69, including a full compliance with Appendix T to 10 CFR Part 50, no prior NRC review and approval is necessary. In this case the licensee notifies the Director of Nuclear Reactor Regulation (NRR) of its intent to implement §50.69 along with a proposed schedule of system implementation and regulations that are being adopted. If a licensee determines does not fully comply with Appendix T to Part 50, a submittal is made to the Director of NRR requesting NRC approval on implementing §50.69 for a select set of regulations that are referenced in §50.69. The submittal would include details of the implementation process or reference to a NRC endorsed guideline, including any exceptions taken to such guidelines. Additional details of the submittal are provided in Appendix B.

Plants that follow this~~e~~ guideline should be able to implement risk-informed regulation with minimal NRC review. Since this guidance is used to effect a change to the plant's licensing basis, it follows the principles of NRC Regulatory Guide 1.174, as follows:

1. Proposed increases in risk, if any, are small and are consistent with the Commission's safety goal policy statement.
2. The process will result in changes that are consistent with defense-in-depth philosophy.
3. The process will result in changes that maintain sufficient safety margins.
4. Performance measurement strategies are used to monitor the change.

The process considers the current regulatory requirements, and all available risk information, to determine categorization and treatment of SSCs. The process is effected through the use of a dedicated panel of plant personnel, the integrated decisionmaking panel (IDP). To implement this process, the licensee must have performed a PRA that estimates core damage frequency (and large early release frequency) due to internally initiated events and internal flooding. All plants have used methods to analyze other important risk contributors, such as seismic risk, fire risk, other external event risks (high winds, tornadoes, aircraft impact, etc.) during power operation, and risk during outage conditions. These methods may involve use of a PRA to quantify these risk impacts, or may involve simplified analyses or qualitative methods. Quantification of non-internal event risk is not a requirement for implementation[2], but would be expected to result in additional benefit.

The process for implementation involves four elements:

1. Selection of scope of SSCs to be addressed
2. Categorization of SSCs into high or low risk significance
3. Determination of special treatment requirements based on categorization
4. Monitoring of implementation

[2] As discussed in NRC Regulatory Guide 1.174, quantification of non-internal event risk may be necessary if the aggregate risk impact exceeds the "very small change" guidelines for CDF and LERF.

The first element involves determining the plant systems to which the revised approach would be applied. Plant systems that can impact PRA initiating events and accident mitigation are candidate systems for application of the process. Certain plant systems have regulatory requirements that have bases other than protection of public health and safety from potential reactor accidents (e.g., the radwaste processing system). These systems, and their associated regulatory requirements, are not within the scope of the process.

The approach may be applied to all candidate systems, or may be applied to selected systems. The preferred approach is to apply the revised categorization and treatment provisions to all candidate systems. Selective implementation will incur complexities resulting from the need to maintain two separate regulatory programs. However, selective implementation may be undertaken provided the application meets the four Reg. Guide 1.174 principles listed above.

The second ~~principal activity~~element is the categorization of the SSCs according to safety significance[3]. Treatment requirements for SSCs will be dependent on this safety classification. This report establishes an integrated process, which relies upon the insights from plant-specific risk analyses and other engineering and operating inputs for use in the categorization of SSCs. The categorization process has been ~~constructed~~ developed to build on the previous risk-informed categorization activities. A licensee is not required to repeat activities that have already been completed as part of a previous risk-informed categorization process.

The necessity of addressing each component, or each part of a component is determined by each licensee based on the anticipated benefit. A licensee may determine that it is sufficient only to perform system or subsystem analyses. In such cases, all the components within the boundaries of the subsystem or system would be governed by the same set of safety-significant functions. Each licensee has the option, based on the estimated benefit, of performing additional engineering and system analyses to identify specific component level or piece part functions and attributes.

The regulatory change process (10 CFR 50.59) focuses on ~~applies only to~~ activities that are directly associated with ~~encompassed by~~ the 10 CFR 50.2 ~~definition of~~ design bases and that are ~~or~~ described in the final safety analysis report. In a risk-informed regulatory environment, management focus should be on operational activities and equipment that have safety significance, which may not necessarily comport with the aspects of the facility described in the final safety analyses report. For example, containment venting is not described in the final safety analysis reports for most BWRs, but may be a risk significant activity for some plants. As a result, Section 50.69 includes a risk-informed change control process for risk-informed SSCs and activities. The guidance for the new change control process for beyond design bases events is included as part of this guidance document in the section on treatment.

[3] It should be noted that the licensee has the flexibility to determine the categorization to the sub-component level. As a result, there could be low safety-significant components within a system and low safety-significant subcomponents within a safety-significant component. Example, a valve pressure boundary may be safety significant, but its active components may be low safety-significant.

A licensee, in its application to adopt a set of regulations under §50.69, would make a commitment to implement the §50.69 categorization process and special treatment requirements in accordance with this guideline. Changes to the SSC categorization process and SSC treatment as described in this guideline would be governed by NEI 99-04, *Guideline for Managing NRC Commitment Changes.*

1.4 REFERENCES

This guidance was developed considering numerous inputs including the current deterministic design basis of the plants, existing regulations, defense-in-depth, preservation of safety margins, and both qualitative and quantitative risk evaluations. This is consistent with the NRC's PRA Policy Statement issued in August, 1995, and the NRC white paper, *Risk-Informed and Performance-Based Regulation*, issued in March, 1999, which states, "...a risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment including the principle of defense-in-depth and the incorporation of safety margins, and performance history are used ..."

Since 1991, the industry and the NRC has developed background documents and guidance for the application of risk-informed applications. Section 9.0 provides a list of references.~~Several of these documents had significant impact on the development of this guidance, including:~~

- ~~EPRI TR-105396, *PSA Applications Guide*,~~
- ~~Regulatory Guide 1.174, *An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*,~~
- ~~NRC SECY 99-256, *Rulemaking Plan For Risk-Informing Special Treatment Requirements*.~~
- ~~NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*~~
- ~~NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*~~
- ~~NRC Regulatory Guides 1.175, 1.176, 1.177, and 1.178, and~~
- ~~ASME Code Case OMN-3, *Requirements for Safety Significance Categorization of Components using Risk Insights for Inservice Testing of LWR Power Plants*.~~

~~Each of these documents recommends the use of an integrated decision process that combines operating experience, engineering analyses, expert opinions, structured qualitative analyses, and quantitative evaluations. The approach described in this guidance is consistent with the processes and approaches described in these referenced documents.~~

Table 1.1
Special treatment regulations subject to optional risk-informed approach of 10 CFR 50.69

50.34, Contents of applications; technical information (FSAR)
50.44, Standards for combustible gas control system in light-water-cooled power reactors
50.49, Environmental qualification
50.54, Conditions of licenses (in reference to Quality Assurance Programs only)
50.55, Conditions of construction permits
50.55a, Codes and standards
50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.
~~50.63, Loss of all alternating current power.~~
50.65, Monitoring effectiveness of maintenance
50.72/50.73, Reporting
Appendix A, General Design Criteria
GDC 1, Quality standards and records
~~GDC 2, Design bases for protection against natural phenomena~~
~~GDC 3, Fire protection~~
~~GDC 4, Environmental and dynamic effects design bases~~
GDC 37, Testing of emergency core cooling system
GDC 40, Testing of containment heat removal system
GDC 42, Inspection of containment atmosphere cleanup systems
GDC 43, Testing of containment atmosphere cleanup systems
GDC 45, Inspection of cooling water system
GDC 46, Testing of cooling water system
Appendix B, Quality Assurance
Appendix J, Containment leakage
~~Appendix R, Fire Protection~~
Appendix S, Seismic
Part 21, Reporting of defects and noncompliance
Part 52, Advanced Reactors
Part 54, License Renewal
Part 100, Appendix A, Seismic

2 CATEGORIZATION PROCESS

2.1 GUIDING PRINCIPLES

Before describing the categorization process, it is useful to understand the objectives that drove the development of the process, and the guiding principles that govern the process and criteria.

The objective of this guidance is to establish the process and criteria for determining the SSCs that require special treatment. By defining the SSCs that require special treatment, those that do not require special treatment are identified by exception. The process and criteria are intended to be sufficiently clear and robust such that if a licensee's program meets the criteria there is no need for prior NRC review and approval of the plant-specific program.

As the process and criteria were developed, a number of guiding principles ~~were~~were used to steer the process. These principles are:

- **Applicable Risk Assessment Information Will Be Utilized**

As a result of the Individual Plant Examination (IPE) program and a number of industry efforts, all licensees have gained an appreciation for the degree of susceptibility to and the performance of their plants under severe accident conditions. The IPE process required the evaluation, at least qualitatively, of the risks during power operations of a spectrum of hazards including internal events, fires, earthquakes, high winds, and floods. Industry initiatives have led to the institution of shutdown safety programs aimed at managing risks during low power and shutdown conditions.

Quantitative probabilistic risk analyses have been performed for at least some of these hazards. In cases where quantitative analyses are not available, at least screening evaluations have been performed. Quantitative analyses are highly amenable to identifying the most (or least) significant SSCs. However, many of the screening analyses, both quantitative and qualitative, can also yield plant specific information which can be used in determining the safety significance of an SSC. For this reason, all available plant-specific risk assessment information is expected to be brought to bear in the categorization process.

- **If No PRA Information Exists Related to A Particular Hazard or Operating Mode, Deterministic or Qualitative Information Will Be Relied Upon**

In cases where PRAs or other quantitative analyses are not available, deterministic or qualitative information will be relied upon. For example, if a plant does not have a tornado risk assessment, then the features of the plant which were designed specifically to protect systems or components from failure during a tornado will be considered safety significant. This may be conservative for some plants. In those cases, the licensee always has the option to perform a risk assessment of the hazard to determine if those SSCs would truly be considered safety significant. As a result, plants with more plant-specific PRA information available may find more SSCs being classified as low safety significant.

- **The Classification Process Should Employ a Blended Approach Considering Both Quantitative PRA Information and Qualitative Information**

  Consistent with the principles of Regulatory Guide 1.174, the implementation of a risk-informed approach includes both the consideration of quantitative information gained the performance of plant specific PRAs and qualitative information regarding defense-in-depth and safety margins.

- **The Principles of the NRC's Risk-Informed Approach to Regulations, As Embodied In Reg. Guide 1.174 Will Be Maintained**

  The risk-informed approach described herein is intended to utilize the principles of the NRC's risk-informed approach to regulation:

  1. The Proposed Change Meets the Regulations - The changes in special treatment will be made under the NRC's proposed 10CFR50.69.

  2. The Proposed Change Is Consistent With The Defense-In-Depth Philosophy - The re-classification and treatment process provides reasonable assurance that safety functions are maintained. Therefore, defense-in-depth will not be impacted. As part of the classification process, a review is performed which assesses the ~~level of~~ role the SSC plays in ensuring defense-in-depth ~~without credit for SSCs defined as low safety significant~~. ~~In addition, the impact of common cause failure of SSCs, which are modeled in a PRA and are classified as low safety significant, is considered in the treatment.~~

  3. The Proposed Change Maintains Sufficient Safety Margins - The reclassification and treatment process provides reasonable assurance that safety functions are maintained. Therefore, safety margins will not be impacted.

  4. Any Increases in Core Damage Frequency or Risk Should Be Small and Consistent With the Intent of the Commission's Safety Goal Policy Statement - The re-classification and treatment process provides reasonable assurance that safety functions are maintained. Risk sensitivity studies will be used to demonstrate that no significant change in CDF and LERF will be expected.

  5. The Impact Of The Proposed Change Should Be Monitored Using Performance Measurement Strategies - Performance monitoring strategies will employed as part of the treatment process.

- **Where an Engineering Basis for Reclassification Can Not Be Developed, No Change in Treatment of the SSC Will Occur**

  As discussed in Section1, it is anticipated that many ~~safety related~~safety-related SSCs ~~will~~would be categorized as RISC-1. Likewise, it is anticipated that many non-~~safety related~~safety-related SSCs will be categorized as RISC-4. An engineering basis, subject to evaluation by an Integrated Decision-making Panel (IDP), is required for selection of other pathways. This engineering basis must be developed from a risk-informed perspective.

- **The Attribute(s) Which Make An SSC Safety Significant Will Be Factored Into Treatment**

  The results of the numerous plant-specific PRAs which have been performed indicate that the attributes of an SSC which make it safety significant may or may not be the same attributes which governed its original ~~safety related~~safety-related classification. For example, some ~~safety related~~safety-related SSCs have functions for beyond design basis events, which were not considered in the original design. BWR containment vent valves are a good example of this. They are generally containment isolation valves designed to ~~assure~~ ensure the containment is isolated in the event of a design basis accident. However, most BWR PRAs would find that the function of opening to allow venting for containment pressure control to be safety significant. In other cases, non-~~safety related~~safety-related SSCs, which were not credited in design basis analyses, are found to be risk significant (e.g., feedwater and condenser in some BWRs, startup feedwater pumps in some PWRs).

  As a result, the categorization process focuses on the attributes that define why an SSC is safety significant. This allows the special treatment requirements to focus on those attributes that are most important.

- **The Treatment For RISC-3 SSCs Will Be Designed to Maintain Function**

  The overall philosophy of the treatment changes for ~~safety related~~safety-related, low safety significant SSCs (RISC-3) is to provide sufficient confidence ~~reasonable assurance~~ that the safety functions will be available. This allows continued confidence that the design basis of the plant can be met and reduces the need to compute any estimated increase in risk due to the change in classification.

2.2 SAFETY SIGNIFICANT ATTRIBUTES OF SSCs

One of the central concepts of the risk-informed safety categorization process is performance attributes. The risk-informed performance of many SSCs is the same (or similar) as that required in the design basis. At one time, it was expected that the design basis attributes would envelope all performance attributes. In many cases, this is true. For example, stroke times for valves are generally set based on conservative thermal hydraulic analyses that lead to performance requirements far in excess of those which a

PRA would require (valves required to open in seconds when the PRA may indicate that minutes are available). In other cases, SSCs can have significantly different performance needs for severe accident mitigation. SSCs may be used in a unique manner or the conditions under which performance is desired may be more severe than the design basis considered. For example, pressurizer PORVs have a design basis to open to relieve primary system pressure. While that function may (or may not) be important in a PRA, another function not considered in the design basis is likely to be: open on demand to support bleed and feed cooling of the RCS in the event of loss of all secondary cooling.

The process described in this guideline addresses this issue by identifying the attributes of SSC performance, which make the SSC safety significant so that the special treatment requirements can be focused on those attributes. Safety-significant functional (performance) attributes are identified for each structure, system, or component based on the SSCs contribution to the safety-significant function.

Functional attributes can be broadly classified into four major categories:

- SSC Function

  Some SSCs perform an entirely different function in severe accident mitigation than their design basis function (e.g., valves required close for design basis, open for severe accidents).

- Performance Attributes

  The function of the SSC is the same, but the SSC is expected to perform in a capacity beyond design basis limits (e.g., containment ultimate pressure capability).

- Environmental Factors

  Some SSCs are credited in PRAs as being capable of operating outside the design basis envelope (e.g., pumps expected to operate without room cooling).

- Actuation Requirements

  Often, due to less stringent performance requirements, some SSCs are credited in PRAs based on manual actuation (e.g., timely manual actuation of injection systems).

2.3 OVERALL APPROACH

The overall approach to the risk-informed categorization process described in this guideline involves a four-step process. Figure 2.3-1 provides an overview of this process.

Figure 2.3-1

Step 1: Assembly of Plant-Specific Risk Information → **Step 2: Compilation of Risk Insights & Safety Significant Attributes** → **Step 3: IDP Review & Classification. Recommend Changes In Treatment** → **Step 4: Evaluation of Recommended Changes**

Resources:
- ✓ Internal Events PRA
- ✓ Fire PRA/FIVE
- ✓ Seismic PRA/Seismic Margins
- ✓ External Events PRA/ IPEEE Screening
- ✓ Shutdown PSA/ Shutdown Safety Management
- ✓ SSC Design Basis Information

Includes:
- ✓ Risk Significance Assessments
- ✓ Integrated Risk Significance Assessment
- ✓ Defense-in-Depth Assessment
- ✓ Identification of Safety Significant Attributes
- ✓ Basis for Low Safety Significance for Safety Related SSCs

Considers:
- ✓ Risk Insights
- ✓ Safety Significant Attributes
- ✓ Operating History
- ✓ Deterministic Considerations
- ✓ Defense-in-Depth
- ✓ Safety Margins

Includes:
- ✓ Qualitative Assessment of Treatment Impacts
- ✓ Risk Sensitivity Studies
- ✓ Evaluation of Monitoring Effectiveness

Risk-Informed Classification Process

Step 1: Assembly of Plant-Specific Risk Information → **Step 2: Compilation of Risk Insights & Safety Significant Attributes** → **Step 3: IDP Review & Classification. Recommend Changes In Treatment** → **Step 4: Evaluation of Recommended Changes**

Resources:
- ✓ Internal Events PRA
- ✓ Fire PRA/FIVE
- ✓ Seismic PRA/Seismic Margins
- ✓ External Events PRA/ IPEEE Screening
- ✓ Shutdown PSA/ Shutdown Safety Management
- ✓ SSC Design Basis Information

Includes:
- ✓ Risk Significance Assessments
- ✓ Integrated Risk Significance Assessment
- ✓ Initial Identification of Safety Significance
- ✓ Identification of Safety Significant Attributes
- ✓ Basis for Low Safety Significance for Safety Related SSCs

Considers:
- ✓ Risk Insights
- ✓ Safety Significant Attributes
- ✓ Operating History
- ✓ Deterministic Considerations
- ✓ Defense-in-Depth
- ✓ Safety Margins

Determines:
- ✓ Specific Changes In Treatment
- ✓ Monitoring

Includes:
- ✓ Qualitative Assessment of Treatment Impacts
- ✓ Risk Sensitivity Studies
- ✓ Evaluation of Monitoring Effectiveness

The first step in the risk-informed categorization process involves the assembly of the relevant plant-specific risk information. In general, as a result of the IPE process and other risk assessment and management activities most utilities have plant-specific analyses in the following areas:

- Internal Events PRA
- Fire PRA/FIVE
- Seismic PRA/Seismic Margins
- External Events PRA/IPEEE Screening
- Shutdown PRA/Shutdown Safety Management

These analyses or programs ~~may represent the current plant design and operation, but even if they have not been kept up to date, they~~ provide insights regarding the plant-specific risk impacts of potential hazards.

The core of the safety significance process is in the second step: Compilation of Risk Insights & Safety Significant Attributes. This involves the evaluation of each structure, system, and component with respect to its safety significance in five hazard areas:

- Internally Initiated Events (including Internal Floods)
- Fires
- Seismic Events
- Other External Events (e.g., tornadoes, high winds, chemical releases, etc.)
- Shutdown operations

These areas correspond to the topical risk analyses (or other assessments) already performed by utilities. This step involves the assessment of SSC risk significance in each of these areas, development of an integrated risk significance across those areas with quantitative assessments, development of initial recommendations on safety significance classification for input to the IDP, identification of the safety significant attributes of SSCs identified as safety significant (i.e., RISC-1 and RISC-2) and development of bases for the low safety significance of ~~safety related~~safety-related SSCs evaluated. This step will be performed largely by personnel familiar with the plant-specific analyses gathered in Step 1 (i.e., the plant PRA group).

The third step in the risk-informed categorization process involves the review of the results of Step 2 by the Integrated Decision-making Panel. The purpose of this panel is to review the risk information developed in Step 2 and evaluate other considerations, which are part of a risk-informed process. The result of the IDP review is the classification of SSCs and identification of the changes in treatment and monitoring. The IDP is a multidisciplinary team of experts that can bring together an understanding of design, operational, licensing, and risk perspectives.

The fourth and final step in the process is the evaluation of the risk sensitivity of the recommended changes. This step involves both qualitative and quantitative assessments of the anticipated impact of the proposed changes. In general, since one of the guiding principles of this process is that changes in treatment should continue to provide sufficient confidence that the design bases functions ~~not degrade performance~~ for RISC-3 SSCs, and the beyond design bases functions for RISC-2 SSCs would be maintained. As a result, ~~expected to maintain or improve in performance, it is anticipated that~~ there would be little, if any, net increase in risk. This assessment involves the review of the specific treatment changes recommended by the IDP to identify the anticipated impact on a qualitative basis. For those cases where some degradation in performance may be possible, sensitivity studies will be performed using available PRA information. Any identified monitoring will also be evaluated to ~~assure~~ ensure that degradations will be identified appropriately. ~~Should~~ If significant risk impacts ~~be~~ are identified, then those would be referred back to the IDP for further evaluation.

Section 2.4 provides a more detailed description of each step of this process.

2.4 SPECIFIC GUIDANCE

This section provides a description of the specific processes and criteria to be applied in the performance of risk-informed safety categorization. The outline of the section follows the four step process described in Section 2.3:

- Assembly of Plant-Specific Risk Information (Sec. 2.4.1)
- Compilation of Risk Insights & Safety Significant Attributes (Sec. 2.4.2)
- IDP Review & Classification. Recommend Changes In Treatment (Sec. 2.4.3)
- Risk Evaluation of Recommended Changes (Sec. 2.4.4)

2.4.1 Assembly of Plant-Specific Risk Information

The first step in the categorization process is the collection and assembly of plant-specific resources that can provide input to the determination of safety significance.

2.4.1.1 Documentation Resources

Like all risk-informed processes, the categorization process relies upon input from both standard design and licensing information, and risk analyses and insights.

The understanding of the risk insights for a specific plant are generally captured in the following analyses:

- Internal Events PRA,
- Fire PRA or FIVE Analysis,
- Seismic PRA or Seismic Margin Assessment,
- External Hazards PRA(s) or IPEEE Screening Assessment of External Hazards, and
- Shutdown PRA or Shutdown Safety Program developed per NUMARC 91-06.

Examples of resources that can provide information on the safety classification and design basis attributes of SSCs include:

- Master Equipment Lists (provides ~~safety related~~safety-related designation)
- UFSAR
- Design Basis Documents
- 10 CFR 50.2 Assessments

2.4.1.2 Use of PRA Information

At a minimum, a PRA modeling the internal initiating events at full power operations must be used to provide input to the IDP. At a minimum, the PRA must be capable of quantifying core damage frequency (CDF) and large early release frequency (LERF) for power operations due to internal events and must reasonably reflect the as-built and as-operated plant. A reasonable reflection of the plant is assumed to exist if the PRA has been updated within the previous two years to reflect design and operating history of important systems/components and significant design and procedural changes and Maintenance Rule a(1) SSCs added since the last update have been reviewed to ~~assure~~ensure that the results and insights are not expected to be affected. Assessments of other hazards and modes of plant operation will be reviewed to ~~assure~~ ensure that the results and/or insights are applicable to the as-built, as-operated plant. PRAs provide an integrated means to assess relative significance. In cases where applicable quantitative analyses are not available, the categorization process will generally identify more SSCs as safety significant than in cases where full scope PRAs are available.

A PRA used in this process should be performed in a manner that is consistent with accepted practices, in terms of the scope and level of detail for the hazards evaluated. One effective approach to ensuring quality is a peer review of the PRA. Industry PRA ~~certification~~ peer review programs, such as NEI 00-02 (Ref. 8)~~, and PRA cross-comparison studies~~ can be used to help ensure appropriate scope, level of detail, and quality of the PRA.

When available, the industry consensus standards on PRA are also an acceptable means to assure PRA quality.

The licensee should ~~assure~~ ensure that documentation exists for the review process, the qualification of the reviewers, the summarized review findings, and resolutions to these findings~~, where applicable~~. Based on the PRA peer review ~~or certification~~ process and on the findings from this process, the licensee should justify why the PRA is adequate for this application in terms of scope and quality. One product of the peer review ~~certification~~ process is a series of grades in a spectrum of technical areas. Areas with low grades should be reviewed and evaluated to assess whether changes in the PRA are necessary.

Consistent with other engineering analyses conducted to justify changes to a plant's licensing basis, quality assurance activities are appropriate for the categorization process. In this regard, it is expected that for traditional engineering analyses (e.g., deterministic engineering calculations) existing provisions for quality assurance (e.g., Appendix B to 10 CFR Part 50, for safety-related SSCs) will apply and provide the appropriate quality needed. Likewise, when a PRA is used to provide insights into the integrated decision-making panel, it is expected that the PRA will have been subject to quality control.

The following, in conjunction with the other guidance contained in this ~~guideline~~document, describes methods acceptable to ensure that the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is of sufficient quality to be used for regulatory decisions:

- Use personnel qualified for the analysis.
-
- Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses (an independent peer review ~~or certification~~ program can be used as an important element in this process).
-
- Provide documentation and maintain records in accordance with accepted practices.
-
- Provide for an independent audit function to verify quality (an independent peer review ~~or certification~~ program can be used for this purpose).
-
- Use procedures that ensure appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decision-making is changed (e.g., licensee voluntary action) or determined to be in error.

Any existing PRA or analysis can be ~~utilized~~ used to support the categorization process, provided it can be shown that the appropriate quality provisions have been met. ~~If the PRA or other analysis has not been updated to reflect all current design and operating conditions, it can still be used as long as the limitations of the study are considered in the initial classification and identified to the IDP for consideration in the final classification.~~

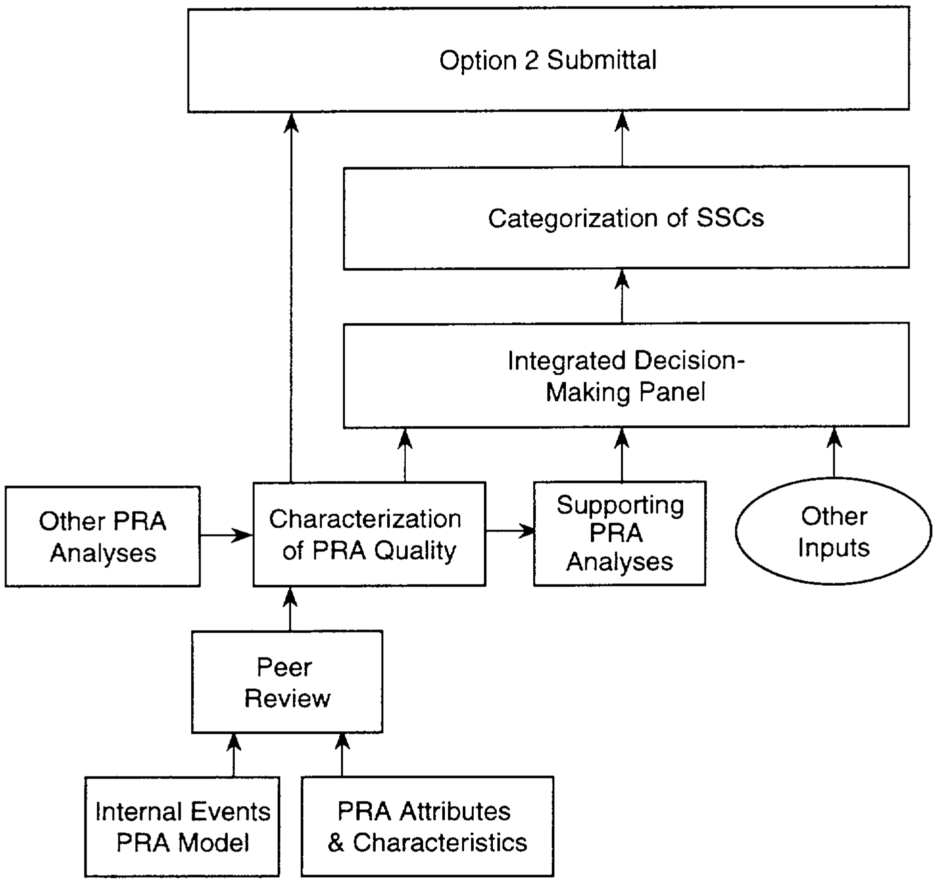
2.4.1.3 Characterization of PRA Quality

~~PRA is a very robust technology which provides unparalleled insight into the role that SSCs play in plant safety. However, like most technologies, PRA has limitations. The figure below defines~~ Figure 2.4-1 depicts the approach to be employed in ~~assuring~~ ensuring the~~at~~ quality of PRA information ~~employed~~ used in the categorization of SSCs.

The primary PRA input into the categorization process is the internal events PRA. This PRA is expected to meet accepted attributes and characteristics and be subject to a peer review. The Industry PRA Peer Review Process (NEI 00-02) represents an acceptable approach to ~~assuring~~ ensuring the quality of the base internal events PRA. The NEI 00-02 peer review provides several ~~outputs which~~outputs, which are useful in characterizing the quality of the PRA. The first output is a set of element grades, ranging from 1 to 4, which provide ~~an~~a consensus assessment by the peer review team of the usability of the PRA in applications. In the terms of the NEI 00-02 grading scheme, the Option 2 categorization process is a Grade 3 application. Thus, elements receiving a grade of 3 or 4 ~~should be~~ are expected to be sufficient to support the categorization process. In cases where a Grade 3 or 4 was achieved through the use of a sensitivity study, the implications of the sensitivity on the categorization process must be assessed. Elements receiving a grade of 1 or 2 should be reviewed by the PRA team to determine the whether the PRA needs to be revised to address the peer review findings or if additional sensitivity studies are called for as part of the categorization process.

Figure 2.4-1

PROCESS FOR ASSURING PRA QUALITY
IN OPTION 2 CATEGORIZATION



The second important output of the NEI 00-02 peer review process are the Fact and Observations (F&Os) ~~which~~ document the strengths and weaknesses of specific aspects of the PRA. F&Os ~~which~~ that identify weaknesses are classified with an importance ranging from A to D, where A is most important and D is generally editorial. All F&Os in categories A and B should be reviewed and dispositioned by either:

- Incorporating appropriate changes into the PRA model prior to use,
- Identifying appropriate sensitivity studies to address the issue identified, or
- Providing adequate justification for the original model, including the applicability of key assumptions to the categorization process.

Other PRA analyses, such as Fire PRAs, Seismic PRAs, and Shutdown PRAs, should be reviewed to ~~assure~~ ensure that (1) none of the internal event peer review ~~certification~~ findings invalidate the results and insights, (2) the study appropriately reflects the as-built, as-operated plant and (3) any new PRA information (e.g., RCP seal LOCA assumptions, physical phenomena, etc.) does not invalidate the results.

The results of the internal events peer review and the review of the other PRA analyses to be used should be summarized in a characterization of PRA quality. This characterization will be provided to the IDP as a basis for the adequacy of the PRA information used in the categorization process and will be summarized in the submittal to the NRC. At a minimum, this characterization should include the following:

Internal Events PRA

- A basis for why the internal events PRA reflects the as-built, as-operated plant.
- A high level summary of the results of the peer review ~~certification~~ of the internal events PRA including elements ~~which~~ that received grades lower than 3.
- The disposition of any peer review fact and observations (F&Os) classified as A or B importance.
- Identification of and basis for any sensitivity analyses necessary to address identified elements and F&Os.

Other PRA Analyses

- A basis for why the other PRA analyses adequately reflect the as-built, as-operated plant.
- A disposition of the impact of the low element grades or serious F&Os on the other PRA analyses.
- Identification of and basis for any sensitivity analyses necessary to address issues identified in the other PRAs.

The Integrated Decision-making Panel (IDP) ~~utilizes~~ uses the PRA quality information, the results of the categorization analyses and other information to recommend a categorization for each SSC. The process to be used by the IDP for the categorization and the justification for adequacy of the PRA information is summarized in the submittal to the NRC.

2.4.2 Compilation of Risk Insights & Safety Significant Attributes

The categorization process described in this section is one acceptable way to undertake the categorization of SSCs. Other methods using a different combination of probabilistic and deterministic approaches and criteria can be envisioned. However, it is expected that the guiding principles (Section 2.1) of this guidance method be maintained. The compilation of risk insights and identification of safety significant attributes builds upon the plant-specific resources. An overview of the safety significance process is shown in Figure 2.4-~~1~~2.

The initial screening is performed at the system/structure level. If the system/structure is found to have a role in a particular portion of the plant's risk profile, then a component level evaluation can be performed.

The first question in the safety significance process involves the role the system/structure plays in the prevention and mitigation of severe accidents. If the system/structure is not involved in severe accident prevention or mitigation, then the screening process is terminated and the assessment of the safety classification is left to the IDP to determine. If all system functions are classified as low safety significant by the IDP, then every component in the system will be classified as low safety significant.

If a system or structure is involved in the prevention or mitigation of severe accidents, then the first risk contributor evaluated is from the internal events PRA. The question of whether a system or structure is evaluated in the internal events PRA (or any of the analyses considered in this guideline) must be answered by considering not only whether it is explicitly modeled in the PRA (i.e., in the form of basic event(s)) but also whether it is implicitly evaluated in the model through operator actions, super components or another aggregated events sometimes used in PRAs. The term "evaluated" means:

- Can it produce a potential initiating event?
- Is it credited for mitigation?
- Is it necessary for another system or structure evaluated in the PRA to prevent an event or mitigate an event?

Some systems and structures are implicitly modeled in the PRA. Personnel knowledgeable in the scope, level of detail, and assumptions of the PRA must make this determination. If the system or structure is determined to be evaluated in the internal events PRA, then the internal event PRA significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.1.

If the system/structure is not evaluated in the internal events PRA, then the assessment of the safety classification relative to internal events is left to the IDP to determine. In either case, the evaluation is continued with fire risk.

If the plant has a fire PRA, then the next step of the screening process is to determine whether the system or structure is evaluated in the fire PRA. This can be an even more difficult assessment to make than for the internal events PRA because of the important (and implicit) role that structures, such as fire barriers play in fire PRAs. Personnel knowledgeable in the scope, level of detail, and assumptions of the fire PRA must make this determination. If the system or structure is determined to be evaluated in the fire PRA, then the fire PRA significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.2.

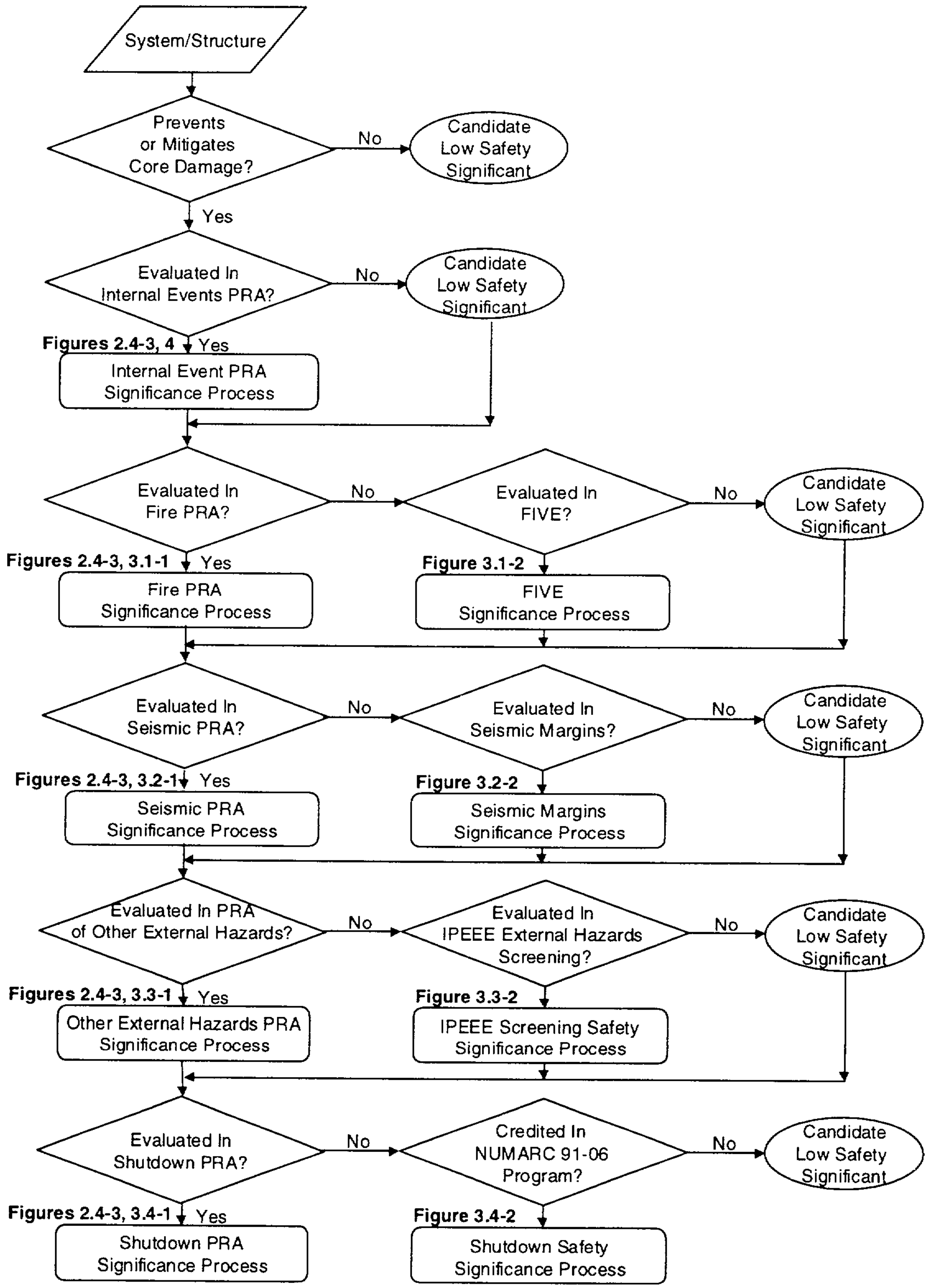
If the plant does not have a fire PRA, then it is likely to have a fire risk evaluation that was performed using the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. Once again, personnel knowledgeable in the scope, level of detail, and assumptions of the FIVE analysis must make this determination. If the system or structure is determined to be evaluated in the FIVE analysis, then the FIVE significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.2.

If the system/structure is not involved in either a fire PRA or FIVE ~~evaluation~~evaluations, then the assessment of the safety classification relative to fire risks is left to the IDP to determine.

Figure 2.4-2

USE OF RISK ANALYSES FOR SSC CATEGORIZATION

~~OVERALL SAFETY SIGNIFICANCE PROCESS~~



If the plant has a seismic PRA, then the next step of the screening process is to determine whether the system or structure is evaluated in the seismic PRA. Often structures are explicitly modeled in seismic PRAs. Personnel knowledgeable in the scope, level of detail, and assumptions of the seismic PRA must make the determination. If the system or structure is determined to be evaluated in the seismic PRA, then the seismic PRA significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.3.

If the plant does not have a seismic PRA, then it is likely to have a seismic margin evaluation that was performed to support the requirements of the IPEEE. Once again, personnel knowledgeable in the scope, level of detail, and assumptions of the seismic margins analysis must make this determination. If the system or structure is determined to be evaluated in the seismic margins analysis, then the seismic margins significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.3.

If the system/structure is not involved in either a seismic PRA or seismic margins evaluation, then the assessment of the safety classification relative to seismic risks is left to the IDP to determine.

If the plant has a PRA, which evaluates other external hazards, then the next step of the screening process is to determine whether the system or structure is evaluated in the external hazards PRA. Often structures are explicitly modeled in external hazards PRAs. Personnel knowledgeable in the scope, level of detail, and assumptions of the external hazards PRA must make the determination. If the system or structure is determined to be evaluated in the external hazards PRA, then the external hazards PRA significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.4.

If the plant does not have an external hazards PRA, then it is likely to have an external hazards screening evaluation that was performed to support the requirements of the IPEEE. Once again, personnel knowledgeable in the scope, level of detail, and assumptions of the external hazards analysis must make this determination. If the system or structure is determined to be evaluated in the external hazards analysis, then the external hazards screening significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.4.

If the system/structure is not involved in either a external hazards PRA or external hazards screening evaluation, then the assessment of the safety classification relative to external hazards risks is left to the IDP to determine.

If the plant has a shutdown PRA, then the next step of the screening process is to determine whether the system or structure is evaluated in the shutdown PRA. Personnel knowledgeable in the scope, level of detail, and assumptions of the shutdown PRA must make the determination. If the system or structure is determined to be evaluated in the shutdown PRA, then the shutdown PRA significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.5.

If the plant does not have a shutdown PRA, then it is likely to have a shutdown safety program developed to support implementation of NUMARC 91-06. Once again, personnel knowledgeable in the scope, level of detail, and assumptions of the NUMARC 91-06 program must make this determination. If the system or structure is determined to be credited in the NUMARC 91-06, then the shutdown safety significance process is used to determine whether it should be considered safety significant for this element of the plant risk profile. This process is discussed in Section 2.4.2.5.

If the system/structure is not involved in a shutdown PRA or NUMARC 91-06, then the assessment of the safety classification relative to shutdown risks is left to the IDP to determine.

~~2.4.1.4~~2.4.2.1 Internal Event Assessment

For systems and structures that are determined to be evaluated in the internal events PRA for the plant, their significance is evaluated using Figures 2.4-~~2~~3 and 2.4-~~3~~4.

The generalized safety significance process for systems and components addressed in a PRA is characterized in Figure 2.4-~~2~~3. This same process is applicable regardless of the scope of the PRA (internal, fire, external, etc.). The first step in this process involves identifying the design basis and severe accident mitigation function(s) ~~which~~ that the system supports. Components within the system are then evaluated to determine whether the ~~PRA required that~~ component was either implicitly or explicitly modeled ~~to perform a safety function evaluated~~ in the PRA (i.e., supported a PRA function). If the component is not modeled~~required~~, then the question of whether it is ~~safety related~~safety-related or not is asked. If it is not ~~safety related~~safety-related, then it is considered a candidate for classification as RISC-4. The term candidate simply refers to the fact that it will be recommended to the IDP for this portion of the risk profile as low safety significant and non-~~safety related~~safety-related. If the component is ~~safety related~~safety-related, but wasn't required to support a PRA function, then before it is preliminarily classified as a candidate RISC-3 component, an investigation is undertaken to determine why it was deemed ~~safety related~~safety-related, but was not required for the PRA.

The importance evaluation can be performed at the system level for the purposes of screening. The remainder of this section discusses the process at the component level, which is the lowest level of detail expected to be performed.

Components, which support a PRA function, are evaluated using the risk importance process shown in Figure 2.4-~~3~~4. Some PRA tools allow for the evaluation of importance measures, which include the role in initiating events. For those cases, the importance measures provide sufficient scope to perform the initial screening. In cases where the importance measures do not include initiating event importance, a qualitative process is

Figure 2.4-3
GENERALIZED SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN PRA

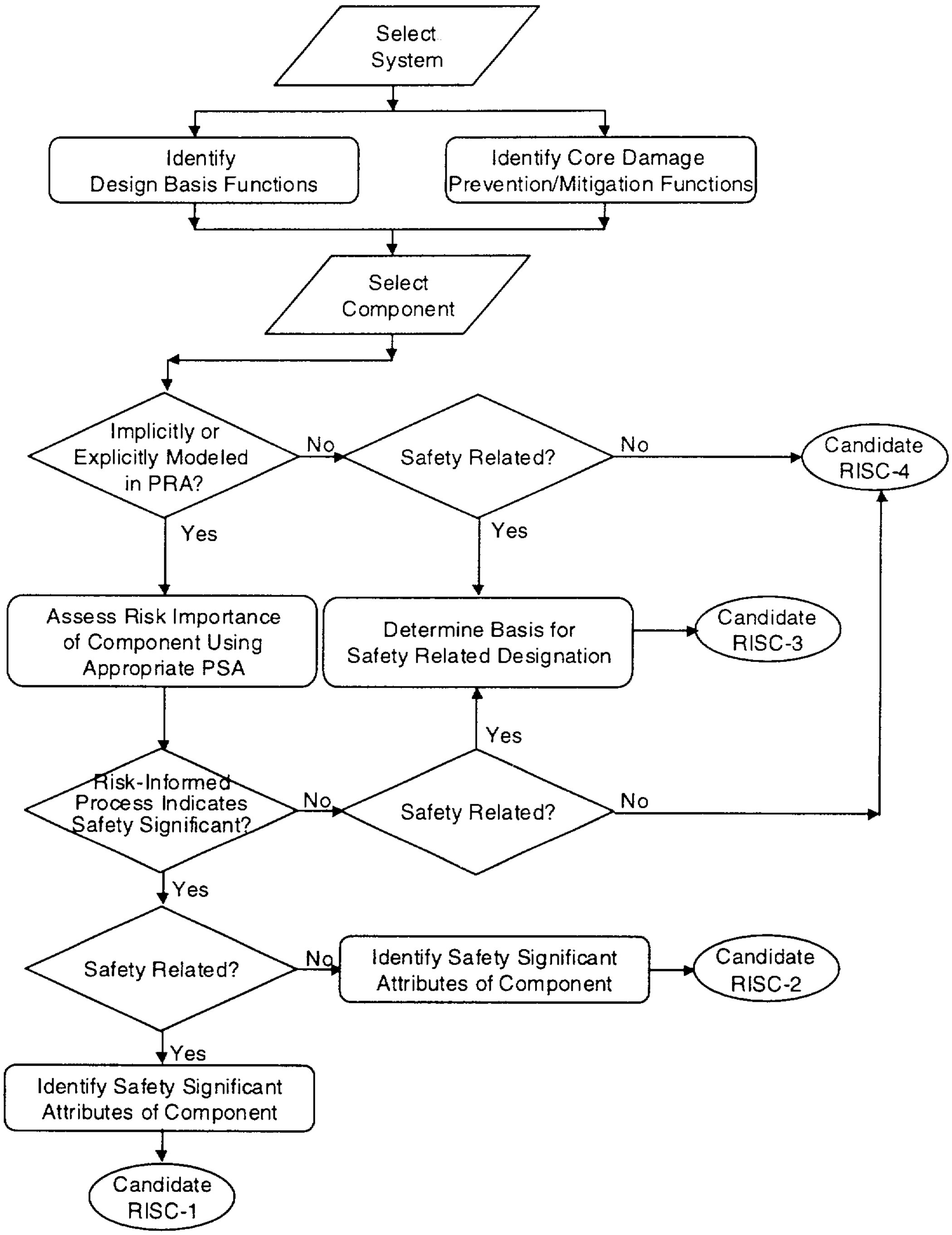
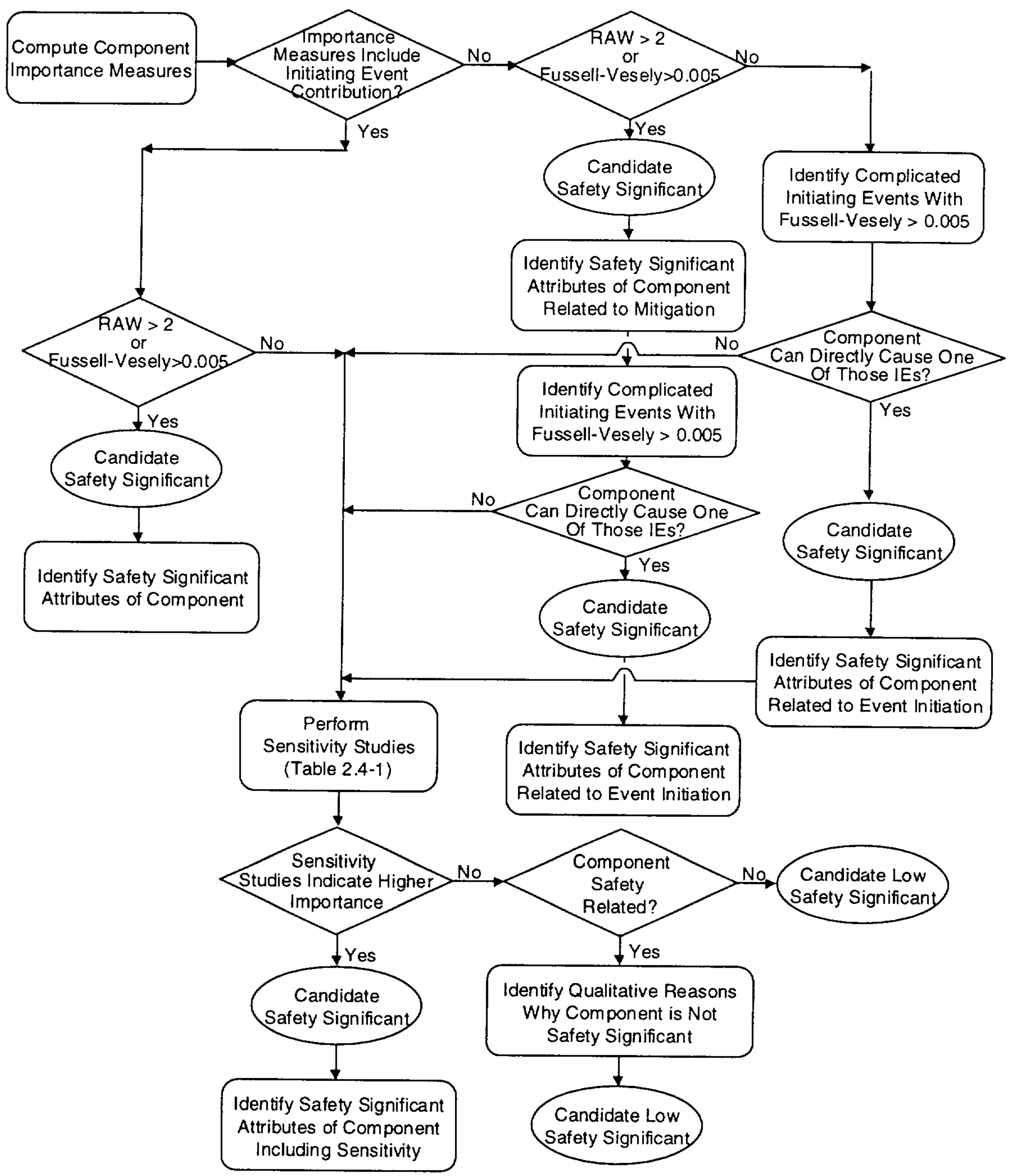


Figure 2.4-4
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN INTERNAL EVENTS AT-POWER PRAs



used to address the initiating event role of the SSC. The mitigation importance of the SSC is assessed using the available importance measures. This process questions whether the SSC can directly cause a complicated initiating event ~~which~~ that has a Fussell-Vesely importance greater than the criteria (0.005). If it does, then it is considered a candidate safety significant SSC and the attributes ~~which~~ that could influence that role as an initiating event are to be identified. A complicated initiating event is considered an event that trips the plant and causes an impact on a key safety function. Examples of complicated initiating events include loss of all Feedwater (PWR/BWR), loss of condenser (BWRs), etc.

The assessment of importance for an SSC involves the identification of PRA basic events ~~which~~that represent the SSC. This can include events ~~which~~ that explicitly model the performance of an SSC (e.g., pump X fails to start), events ~~which~~that implicitly model an SSC (e.g., some human actions, initiating events, etc.) or a combination of both types of events. Personnel familiar with the PRA will have to identify the events in the PRA ~~which~~that can be used to represent each SSC. In general, PRAs are not as capable of easily assessing the importance of passive components such as pipes and tanks. However, in some cases, focused calculations or sensitivity studies can be used. For obtaining risk insights from the PRA for passive pressure boundary components, additional guidance is provided in ASME Section XI, Risk-Informed Safety Classification for Use in Risk-Informed Repair and Replacement Activities, which is under development. Guidance for categorization (and special treatment) for inservice inspection of passive pressure boundary piping components can be obtained from ASME Code Cases N-577 and N-578, along with Westinghouse Owners Group Topical Report WCAP-14572, Revision 1-NP-A and Electric Power Research Institute Report TR-112657 Rev.B-A, respectively.

The risk importance process utilizes two standard PRA importance measures, risk achievement worth (RAW) and Fussell-Vesely (F-V), as screening tools to identify candidate high safety significant SSCs. Risk reduction worth (RRW) is also an acceptable measure in place of Fussell-Vesely. The Fussell-Vesely criteria can be readily converted to RRW criteria. The Fussell-Vesely importance of a component is considered to be the sum of the F-V importances for the relevant failure modes of the component, including common cause failure. The relevant failure modes of a component are ~~those which are expected to be~~those that are expected to be affected by the special treatment requirements being evaluated. If a component does not have a common cause event to be included in the computation of importances, then an assessment should be made as to whether a common cause event should be added to the model. The RAW importance of a component is considered the maximum of the RAW values computed for basic events involving the component. In the case of RAW, the common cause event is not considered in the assessment of component risk significance[4]. The RAW for common

[4] The potential implication of common cause failures introduced by changes in treatment could be evaluated by computing the risk increase assuming all random failures were assumed to be common cause (set the common cause event probability equal to the failure probability of a single component). However, as long as the conditional probability of the common cause failure is greater than 0.005, the F-V importance provides a bounding assessment. That is, a relative risk increase of a factor of 2 can only exist for an event with a F-V of less

cause events is an unrealistic parameter since it reflects the relative increase in CDF/LERF that would exist if a common cause failure condition existed for an entire year.

For example, a motor operated valve may have a number of basic events associated with it, each of which has a separate Fussell-Vesely importance. Likewise, the risk achievement worth of a component is the maximum value determined from the relevant failure modes (basic events):

EXAMPLE IMPORTANCE SUMMARY

COMPONENT FAILURE MODE	**F-V**	**RAW**
1) Valve 'A' Fails to Open	0.002	1.7
2) Valve 'A' Fails to Remain Closed	0.00002	1.1
3) Valve 'A' In Maintenance (Closed)	0.0035	1.7
4) Common Cause Failure of Valves 'A' & 'B' to Open	0.004	n/a
Component Importance	0.00952	1.7
Criteria	> 0.005	>2
Candidate Risk Significant?	**Yes**	

In the above example, Valve 'A' would be considered candidate safety significant due to the total Fussell-Vesely exceeding the criteria. The RAW ~~criteria was~~criteria were not met. The component failure ~~mode which contribute significantly to the importance of Valve 'A'~~mode, which contributes significantly to the importance of Valve 'A', is failure to open (modes 1, 3 and 4). This failure mode is used in the identification of safety significant attributes. If an individual failure mode had not alone exceeded the screening criteria, then the dominant failure mode would be used in defining the attributes.

~~SSCs which~~SSCs, which have high failure probabilities (usually indicative of screening values) and meet the screening criteria solely on the basis of Fussell-Vesely importance, should be identified as candidate safety significant, but the reasons for this classification should be identified to the IDP. In many cases, special treatment will have little or no impact on such SSCs. If the IDP determines that this is the case, it may decide to classify the SSC as low safety significant.

In cases where the internal events core damage frequency is dominated by flooding, it is appropriate to break the evaluation of importance measures into two steps. The first step uses importance measures computed using the entire internal events PRA. The second

than 0.005 if the probability of the event is increased by more than a factor of 200 (1/0.005). Since current common cause methods generally yield conditional probabilities of between 0.1 and 0.01, the use of such a sensitivity is not deemed necessary. However, the conditional probabilities used in the PRA should be reviewed. In cases where values less than 0.005 are used, if the combination of F-V and conditional probability would yield a risk increase of more than 2 would, the SSC should be identified as potentially safety significant.

step uses importance measures computed without the dominant contributor included. This prevents "masking" of importance by the dominant contributor.

If the screening criteria are met for either importance measure, the SSC ~~it~~ is considered a candidate safety significant component and the safety significant attributes are to be identified. If the risk importance measure criteria are not met, then it is not automatically low safety significant. It must be evaluated as part of several sensitivity studies, determined to be low safety significant for all risk contributors and must be reviewed by the IDP. If the importance measures computed by the PRA tool do not indicate that a component meets the Fussell-Vesely or RAW criteria, then sensitivity studies are used to determine whether other conditions might lead to the component becoming safety significant. The recommended sensitivity studies for internal events PRA are identified in Table 2.4-1.

Table 2.4-1
Sensitivity Studies For Internal Events PRA

Sensitivity Study
• Increase all human error basic events to their 95^{th} percentile value
• Decrease all human error basic events to their 5^{th} percentile value
• Increase all component common cause events to their 95^{th} percentile value
• Decrease all component common cause events to their 5^{th} percentile value
~~Increase all component random failure events to their 95th percentile value~~
~~Decrease all component random failure events to their 5th percentile value~~
• Set all maintenance unavailability terms to 0.0
• Any applicable sensitivity studies identified in the characterization of PRA Quality (Section 2.4.1.3)

The sensitivity studies on human error rates, common cause failures, and maintenance unavailabilities are performed to ensure that assumptions of the PRA are not masking the importance of an SSC. The sensitivity addressing the variation of random failure probabilities is performed to ensure that anticipated variations in individual SSC performance would be unlikely to change the classification. In cases where plant-specific uncertainty distributions are not available, other PRAs should be reviewed to identify appropriate parameter ranges. Experience with plant-specific PRAs has shown that the ~~variations in distributions is~~ variations in distributions are relatively small, especially with respect the ratio of the mean and 95^{th} percentile values in lognormal distributions (the most common distribution used in PRAs).

If the sensitivity studies identify that the component could be safety significant, then the safety significant attributes ~~which~~ that yielded that conclusion should be identified.

If, following the sensitivity studies, the component is still found to be low safety significant and it is ~~safety related~~safety-related, it is a candidate for RISC-3. In this case the analyst is ~~expected~~ to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

This risk importance process is performed for both CDF and LERF. In calculating the FV risk importance measure, it is recommended that a CDF (or LERF) ~~cutset~~ truncation level of at least five orders of magnitude below the baseline CDF (or LERF) value be used for linked fault tree PRAs. For example, if the internal events, full power CDF baseline value is 1E-5 /yr, a truncation level of at least 1E-10 /yr is recommended. In addition, the truncation level used should support an overall CDF/LERF which has converged. For linked event tree PRAs, the unaccounted for frequencies should be sufficiently low as to provide confidence that the overall CDF/LERF and resulting importance measures are accurate. When the RAW risk importance measure is calculated by a full re-solution of the plant PRA model, then the truncation level does not significantly affect the RAW calculations. In this case, a default truncation value of 1E-9 /yr seems reasonable. However, if a pre-solved set of cutsets is used to calculate RAWs, the truncation level should be set to a sufficiently low value so that all SSCs with RAW>2 are identified (e.g., cutoff of 1E-10 /yr or lower). The truncation of the PRA model should be checked to ensure that the CDF ~~&~~ and LERF values have converged and that the importance measures are stabilized.

The output of the risk importance evaluation feeds back into the safety significance process of Figure 2.4-2. If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-relatedsafety-related component, then the basis for that designation is questioned and the component is designated as candidate RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

3 CATEGORIZATION OTHER HAZARDS

3.1 FIRE ASSESSMENT

The fire safety significance process takes one of two forms. For plants with a fire PRA, the process is similar to that described for an internal events PRA. This process is shown on Figures 2.4-3~~2~~ and 3.1-1~~2.4-4~~, and is discussed below. Plants ~~which~~that relied upon a FIVE analysis to assess fire risks for the IPEEE would use a modified process shown in Figure 3.1-2~~2.4-5~~7.

The generalized safety significance process for plants with a fire PRA is the same as the process for an internal events PRA. The risk importance process is slightly modified to consider the fact that most fire PRAs do not have the ability to aggregate the mitigation importance of a component with the fire initiation contribution. For that reason, components are evaluated using standard importance measures for their mitigation capability -only~~and separately for the potential to initiate a fire~~. Aside from that small change, the process is the same as the internal events PRA process.

Fire suppression systems which are evaluated using the fire risk analysis can be categorized using this process. In general, fire barriers would not be considered, unless the fire risk analysis supports consideration of the impacts of failure of the barrier. In cases where the impact of fire barrier failure can be evaluated in the risk analysis, the categorization process is applicable.

However, if the fire PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the fire PRA can be considered low safety significant from a fire perspective.

The recommended sensitivity studies for fire PRA are identified in Table ~~2.4-2~~3.1-1.

Table ~~2.4-2~~3.1-1
Sensitivity Studies For Fire PRA

Sensitivity Study
• Increase all human error basic events to their 95th percentile value
• Decrease all human error basic events to their 5th percentile value
• Increase all component common cause events to their 95th percentile value
• Decrease all component common cause events to their 5th percentile value
~~Increase all component random failure events to their 95th percentile value~~
~~Decrease all component random failure events to~~

~~their 5th percentile value~~

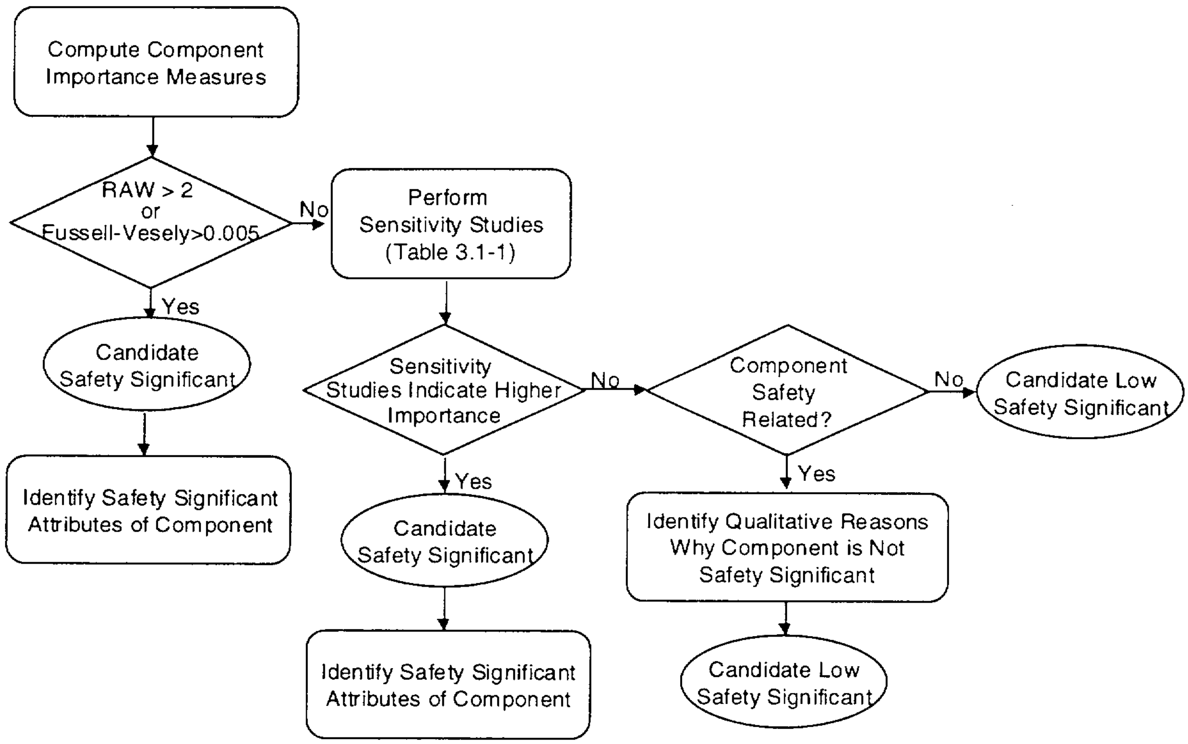
- Set all maintenance unavailability terms to 0.0
- Any applicable sensitivity studies identified in the characterization of PRA Quality (Section 2.4.1.3)
- All manual suppression =1.0
- Any applicable sensitivity studies identified in the characterization of PRA Quality (Section 2.4.1.3)

If the sensitivity studies identify that the component could be safety significant, then the safety significant attributes which yielded that conclusion should be identified. If, following the sensitivity studies, the component is still found to be low safety significant and it is ~~safety related~~safety-related, it could be a candidate for RISC-3. In this case, the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

This risk importance process is performed for both CDF and LERF. Where LERF can not be quantitatively linked into the fire model, the insights from the internal events LERF model should be qualitatively coupled with the assessment of fire impacts on containment isolation to develop recommendations for the IDP on LERF contributors.

The output of the fire risk importance evaluation feeds back into the safety significance process of Figure 2.4-3~~2~~. If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-related component, then the basis for that designation is questioned and the component is designated as a candidate for RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

Figure 3.1-1~~2.~~ ~~4-6~~
RISK IMPORTANCE PROCESS FOR COMPONENTS ADDRESSED IN FIRE PRAs

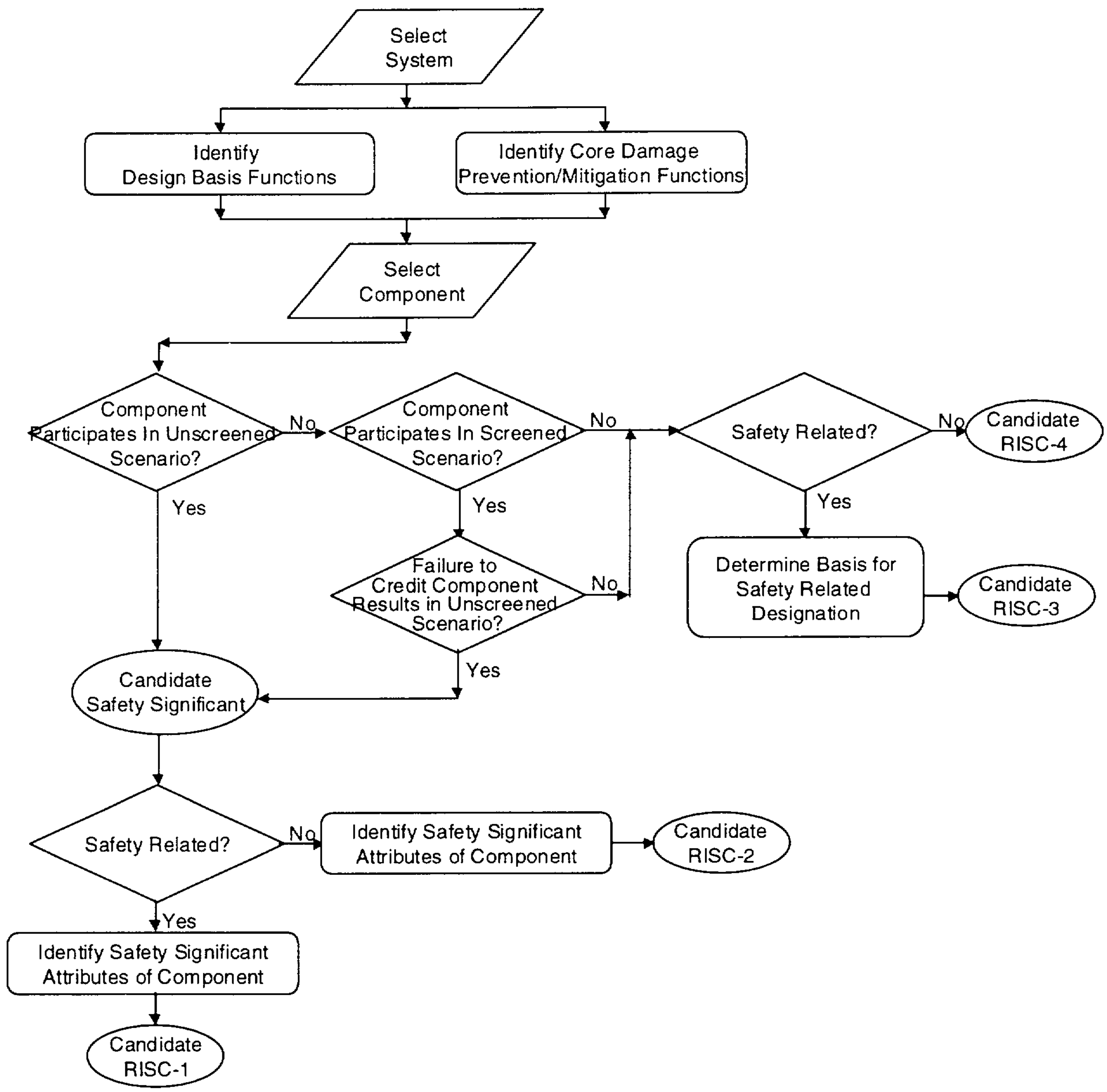


The FIVE methodology is a screening approach to evaluating fire hazards. It does not generate numbers~~, which~~, which are true core damage values; rather, it simply assists in identifying potential fire susceptibilities and vulnerabilities. For this reason, it is somewhat limited in being able to support the identification of low safety significant components. The safety significance process for plants with FIVE evaluations is shown in Figure 3.1-2~~2.4-5~~7.

In this process, after identifying the design basis and severe accident functions of the component, the results of the FIVE analysis are reviewed to determine if any SSCs can be identified as high or low safety significant. If a component participates, either by initiating or in the mitigation of an unscreened fire scenario, it is considered safety significant. This is somewhat conservative since the FIVE process does not generate core damage frequency values. However, the option always exists for the licensee to extend their FIVE analysis to a fire PRA to remove any conservatisms.

If the component does not participate in an unscreened scenario, then its participation in screened scenarios is questioned. If it can be shown that the component either did not participate in any screened scenarios or, even if credit for the component was removed, the screened scenario would not become unscreened, then it is considered candidate low safety significant.

Figure ~~2.4-7~~3.1-2
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN FIVE



~~3.3~~3.2 SEISMIC ASSESSMENT

The seismic safety significance process ~~also~~ takes one of two forms. For plants with a seismic PRA, the process is similar to that described for an internal events PRA. This process is shown on Figures 2.4-3~~2~~ and 3.2-1~~2.4-6~~ and discussed below. ~~For p~~Plants, ~~which~~that relied upon a seismic margins analysis to assess seismic risks for the IPEEE~~,~~ ~~they~~ would use ~~a~~ the modified process shown in Figure 3.2-2~~2.4-7~~9.

The generalized safety significance process for plants with a seismic PRA is the same as the process for an internal events PRA. The risk importance process is slightly modified to consider the fact plant components can not initiate seismic events. Aside from that small change, the process is the same as the internal events PRA process.

However, if the seismic PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the seismic PRA can be considered low safety significant from a seismic perspective.

The recommended sensitivity studies for seismic PRA are identified in Table 3.2-1~~2.4-3~~:

Table 3.2-1~~2.4-3~~
Sensitivity Studies For Seismic PRA

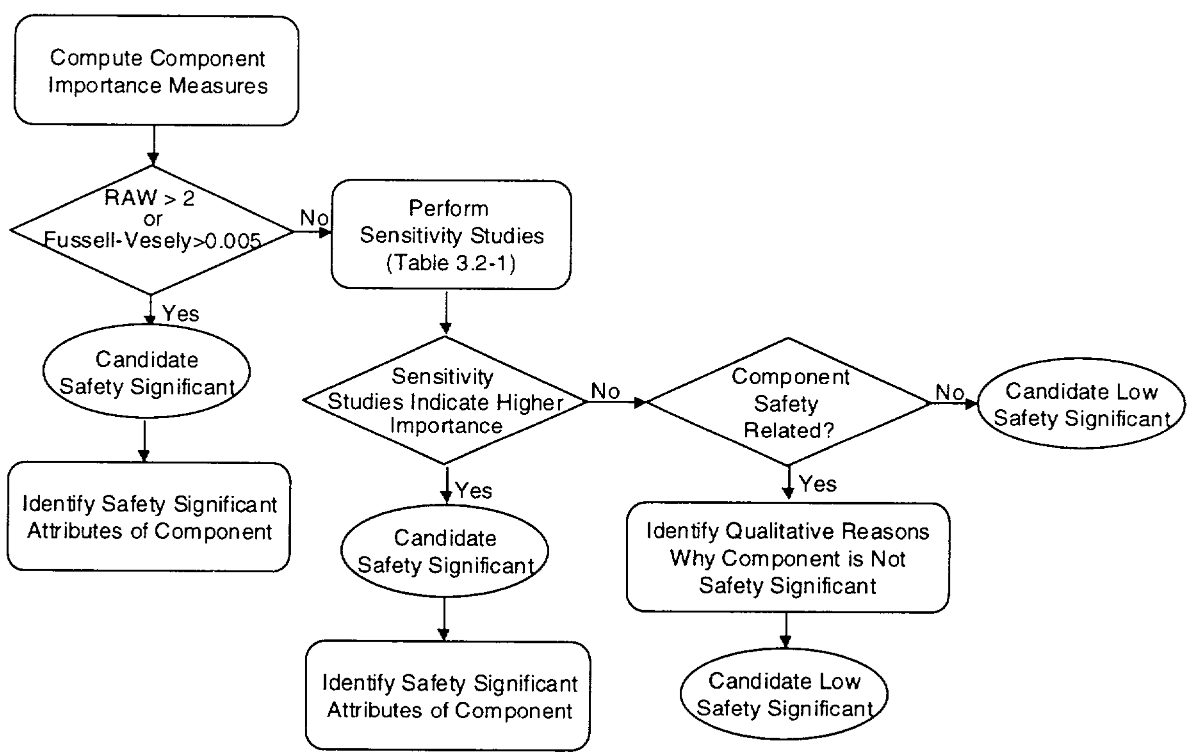
Sensitivity Study
• Increase all human error basic events to their 95th percentile value • Decrease all human error basic events to their 5th percentile value • Increase all component common cause events to their 95th percentile value • Decrease all component common cause events to their 5th percentile value ~~Increase all component random failure events to their 95th percentile value~~ ~~Decrease all component random failure events to their 5th percentile value~~ • Set all maintenance unavailability terms to 0.0 • Use correlated fragilities for all SSCs in an area • Any applicable sensitivity studies identified in the characterization of PRA Quality (Section 2.4.1.3)

If the sensitivity studies identify that the component could be safety significant, then the safety significant attributes which yielded that conclusion should be identified. If, following the sensitivity studies, the component is still found to be low safety significant and it is ~~safety related~~safety-related, it could be a candidate for RISC-3. In this case, the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

This risk importance process is performed for both CDF and LERF. Where LERF can not be quantitatively linked into the seismic model, the insights from the internal events LERF model should be qualitatively coupled with the assessment of seismic impacts on containment to develop recommendations ~~to~~ for the IDP on LERF contributors.

The output of the seismic risk importance evaluation feeds back into the safety significance process of Figure 2.4-~~2~~3. If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-related component, then the basis for that designation is questioned and the component is designated as a candidate for RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

Figure 3.2-1~~2.4-8~~
RISK IMPORTANCE ASSESMENT FOR COMPONENTS ADDRESSED IN SEISMIC PRAs



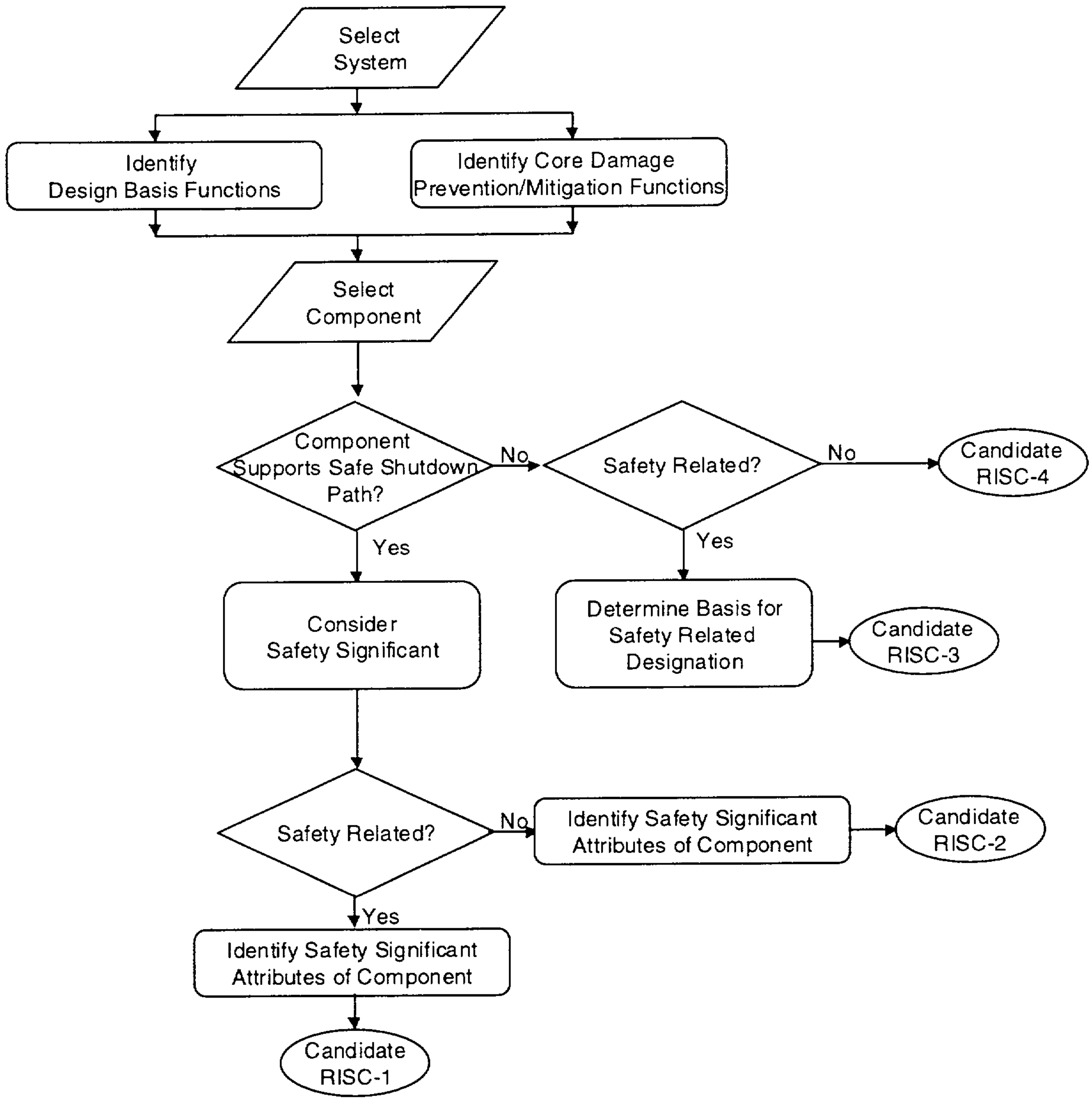
The seismic margins methodology is a screening approach to evaluating seismic hazards. It does not generate core damage values; rather, it simply assists in identifying potential seismic susceptibilities and vulnerabilities. For this reason, it is somewhat limited in being able to support the identification of low safety significant components. The safety significance process for plants with seismic margins evaluations is shown in Figure 3.2-2 ~~2.4-7~~.

In this process, after identifying the design basis and severe accident functions of the component, the seismic margins analysis is reviewed to determine if the component is credited as part of the safe shutdown paths evaluated. If a component is credited, it is considered safety significant. This is conservative since the seismic margin process does not generate core damage frequency values. However, the option always exists for the licensee to perform a seismic PRA to remove any conservatisms.

If the component does not participate in the safe shutdown path, then it is considered a candidate low safety significant with respect to seismic risk.

If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-related component, then the basis for that designation is questioned and the component is designated as a candidate for RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

Figure <ins>3.2-2</ins>~~2.4-9~~
SAFETY SIGNIFICANCE PROCESS FOR
SYSTEMS AND COMPONENTS ADDRESSED IN SEISMIC MARGINS



~~3.4~~3.3 OTHER EXTERNAL HAZARD ASSESSMENT

The significance process for other external hazards (i.e., excluding fire and seismic) also takes one of two forms. For plants with an external hazards PRA, the process is similar to that described for an internal events PRA. This process is shown on Figures 2.4-~~2~~3 and 3.3-1 ~~2.4-8~~ and discussed below. Plants~~, which~~that relied upon an external hazard screening to assess external hazards for the IPEEE~~,~~ would use the ~~a~~ modified process shown in Figure 3.3-2 ~~2.4-9~~.

The generalized safety significance process for plants with an external hazard PRA is the same as the process for an internal events PRA. As for seismic risk, the risk importance process is slightly modified to consider the fact plant components can not initiate external events such a floods, tornadoes, and high winds. Aside from that small change, the process is the same as the internal events PRA process.

However, if the external hazards PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the external hazards PRA can be considered low safety significant from an external hazards perspective.

The recommended sensitivity studies for other external hazard PRAs are identified in Table ~~2.4-4~~3.3-1.~~:~~

Table 3.3-1~~2.4-4~~
Sensitivity Studies For Other External Hazard PRA

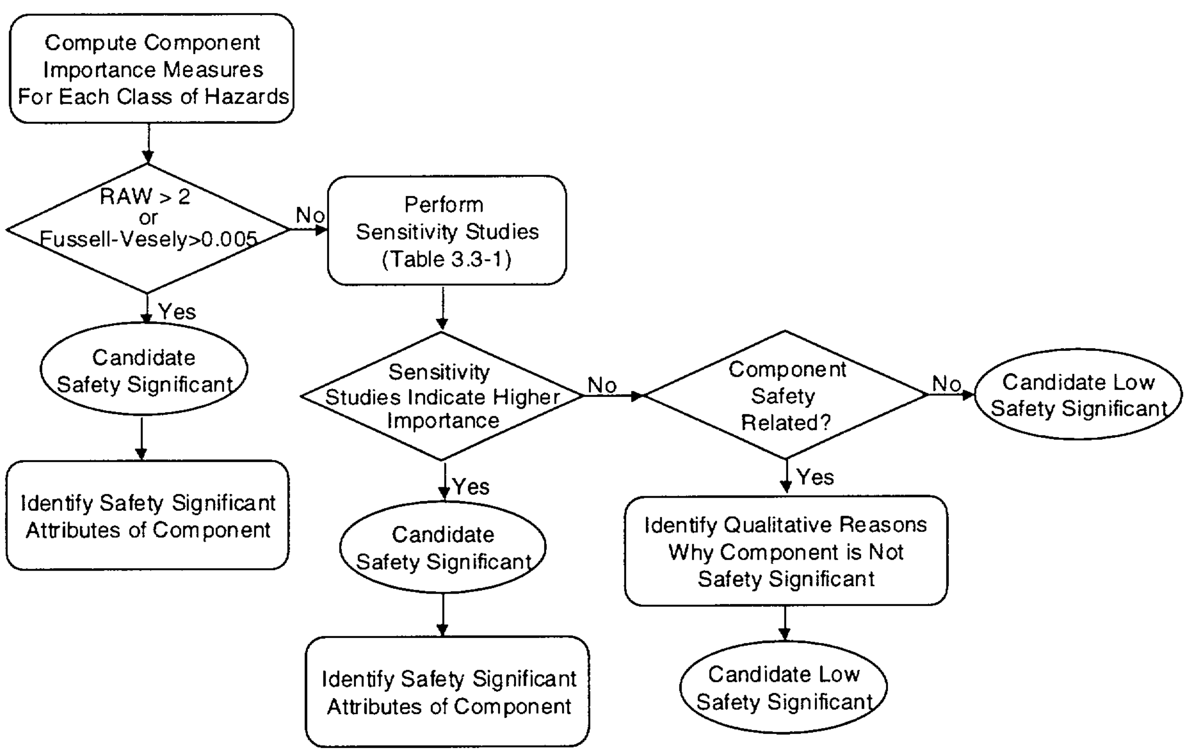
Sensitivity Study
• Increase all human error basic events to their 95th percentile value
• Decrease all human error basic events to their 5th percentile value
• Increase all component common cause events to their 95th percentile value
• Decrease all component common cause events to their 5th percentile value
~~Increase all component random failure events to their 95th percentile value~~
~~Decrease all component random failure events to their 5th percentile value~~
• Set all maintenance unavailability terms to 0.0
• Any applicable sensitivity studies identified in the characterization of PRA Quality (Section 2.4.1.3)

If the sensitivity studies identify that the component could be safety significant, then the safety significant attributes which yielded that conclusion should be identified. If, following the sensitivity studies, the component is still found to be low safety significant and it is ~~safety related~~safety-related, it could be a candidate for RISC-3. In this case, the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

This risk importance process is performed for both CDF and LERF. Where LERF can not be quantitatively linked into the external hazard model, the insights from the internal events LERF model should be qualitatively coupled with the assessment of external hazard impacts on containment to develop ~~a~~recommendations ~~to~~ for the IDP on LERF contributors.

The output of the external hazard risk importance evaluation feeds back into the safety significance process of Figure 2.4-~~2~~3. If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-related component, then the basis for that designation is questioned and the component is designated as a candidate for RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

Figure 3.3-1~~2.4-10~~
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN EXTERNAL EVENTS PRAs



The external hazard screening does not generate core damage values; rather it simply assists in identifying that the plant has no significant external hazard susceptibilities and vulnerabilities. For this reason, it is somewhat limited in being able to support the identification of low safety significant components. The safety significance process for plants with external hazard screening evaluations is shown in Figure 3.3-2~~2.4-9~~.

In this process, after identifying the design basis and severe accident functions of the component, the external hazard analysis is reviewed to determine if the component is credited as part of the safety shutdown paths evaluated. If a component is credited, it is considered safety significant. This is conservative since the external hazard screening process does not generate core damage frequency values. However, the option always exists for the licensee to perform an external hazard PRA to remove any conservatisms.

The process of assessing whether an SSC is safety significant due to other external hazards is as follows:

1) Identify a safe shutdown path for each external event challenge (presumably the same as the seismic shutdown path).

2) The NEI 00-04 screening approach is then to:

   a) Review the SRP on the NUREG 1407 analysis to determine if the SSC is credited as part of the identified safe shutdown path.

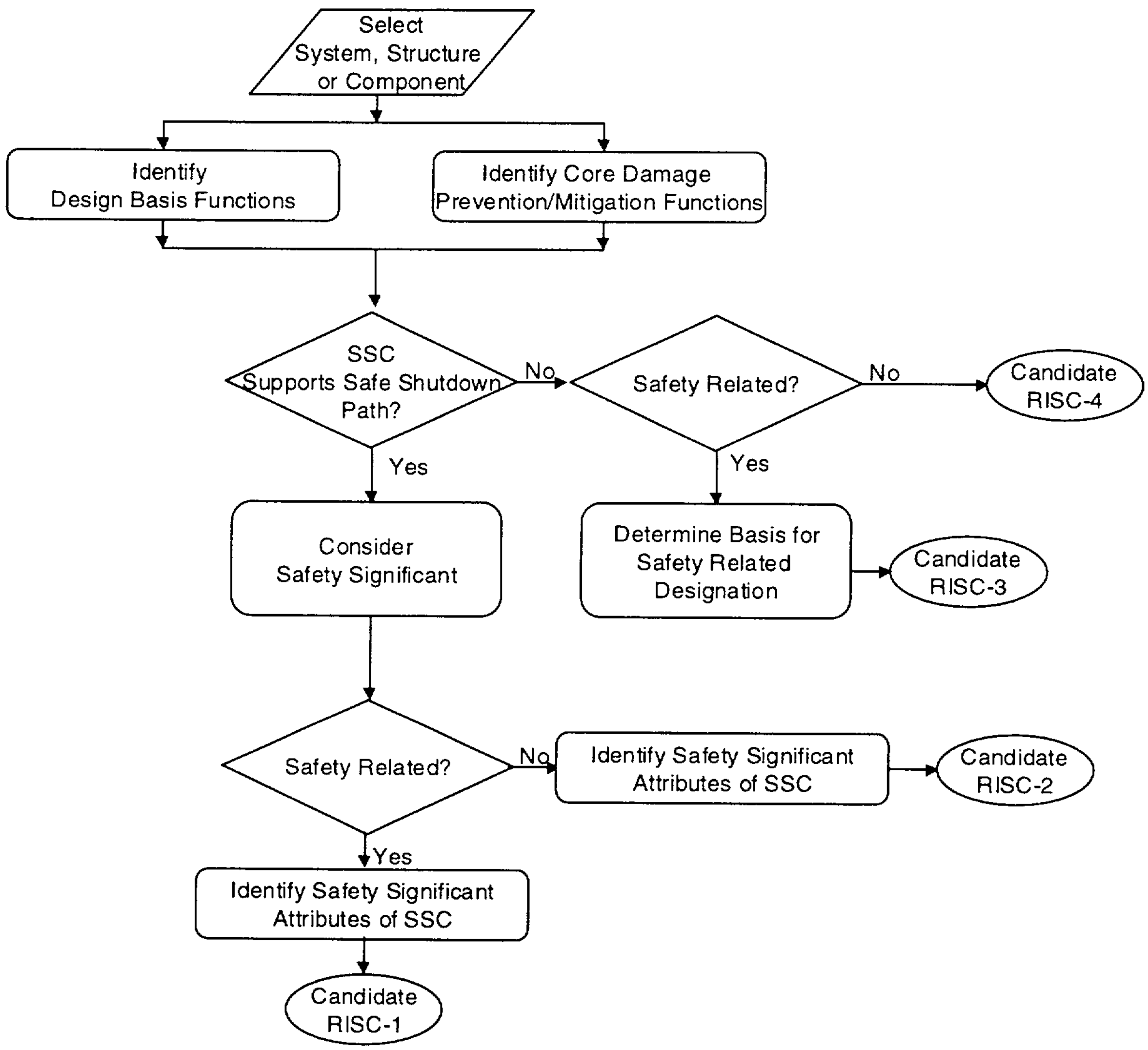
      If a component is credited, it is considered safety significant.

   b) Ensure that the SSC is not relied upon to support or protect any of the SSCs supporting safe shutdowns functions given the challenges to the SSC resulting from the "other" external event. If a component is credited to be available under these conditions, it is considered safety significant, as are the SSCs which assure the functionality of those safety significant SSCs.

If the SSC passes these screens, then the answer to the question "SSC Supports Safe Shutdown Path?" can be "no."

If the component does not participate in the safe shutdown path, then it is considered a candidate low safety significant with respect to external hazards.

Figure 3.3-2~~2.4-11~~
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN EXTERNAL EVENT SCREENING



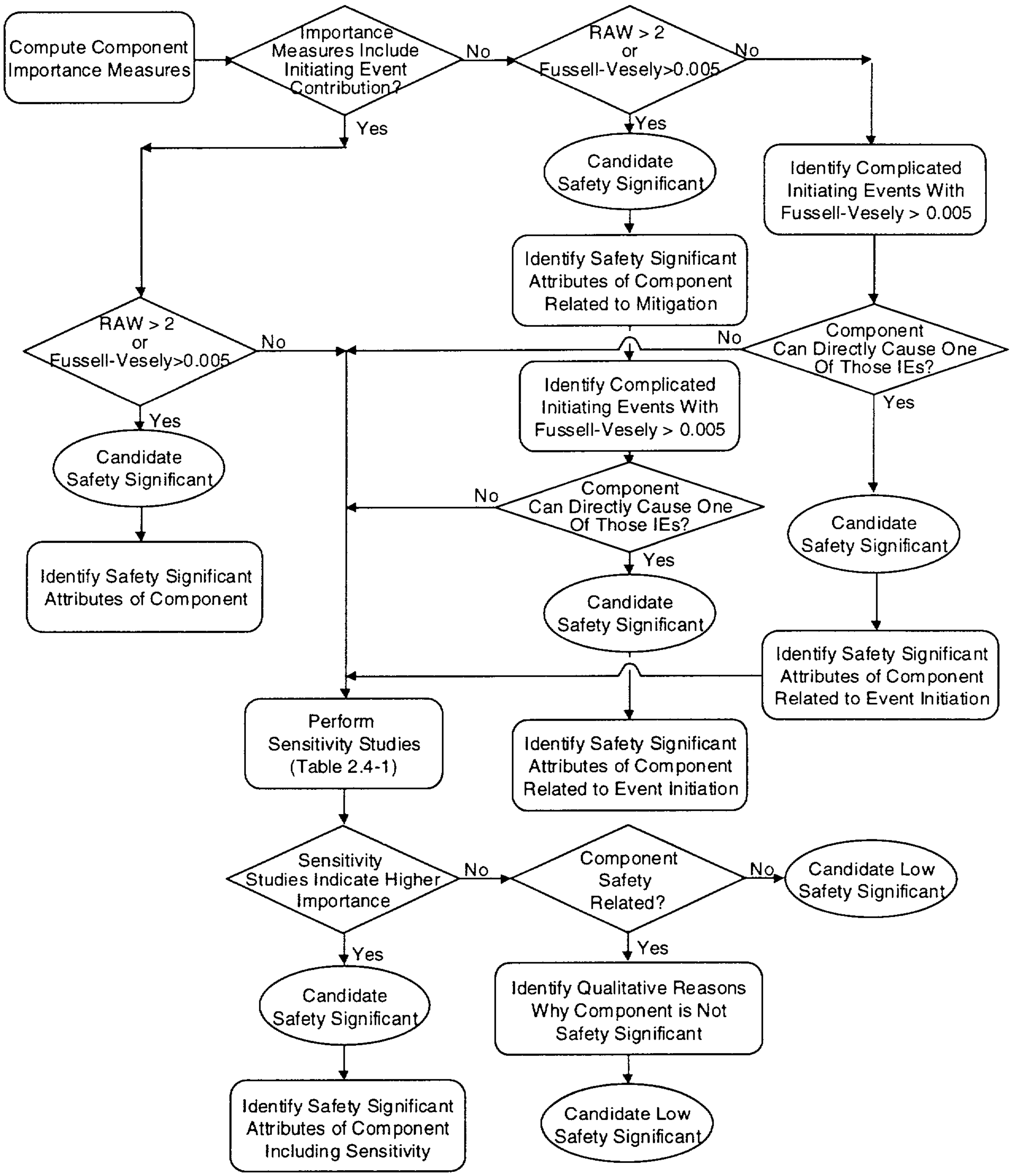
3.4 SHUTDOWN ASSESSMENT

The shutdown safety significance process also takes one of two forms. For plants with a shutdown PRA that is comparable to an at-power PRA (i.e., generates annual average CDF/LERF), the process is similar to that described for an internal events PRA. This process is shown on Figures 2.4-3~~2~~ and 3.4-1~~2.4-10~~. Plants~~, which~~that do not have a shutdown PRA~~,~~ would use ~~a~~ the modified process shown in Figure 3.4-2 ~~2.4-11~~ based on their NUMARC 91-06 program. Due to the similarities between shutdown and at-power PRAs, the generalized safety significance process for plants with a shutdown PRA is the same as the process for an internal events PRA.

However, if the shutdown PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the shutdown PRA can be considered low safety significant from a shutdown perspective.

The same sensitivity studies identified in Table 3.4-1~~2.4-4~~ should be used in the evaluation of shutdown risk significance.

Figure 3.4-1~~2.4-12~~
RISK IMPORTANCE ASSESSMENT PROCESS
FOR COMPONENTS ADDRESSED IN
LOW POWER/SHUTDOWN PRAs
(Same as Internal Events PRA)



Meeting the guidelines for shutdown safety identified in NUMARC 91-06 is not equivalent to a shutdown PRA and does not generate quantitative information comparable to core damage values. Rather, it simply attempts to ensure that the plant has an appropriate complement of systems available at all times. The safety significance process for plants without a shutdown PRA is shown in Figure 3.4-2~~2.4-11~~.

In this process a component can be identified as safety significant for shutdown conditions for one of two reasons:

1. It could initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.),

2. It satisfies both of the following conditions:

   - It participates in a safety function whose failure can result in increasing CDF or LERF, and
   - The minimum requirements[5] cannot be met for the safety function without the system, structure, or component.

If the component does not participate in either of these manners, then it is considered a candidate as low safety significance with respect to shutdown safety.

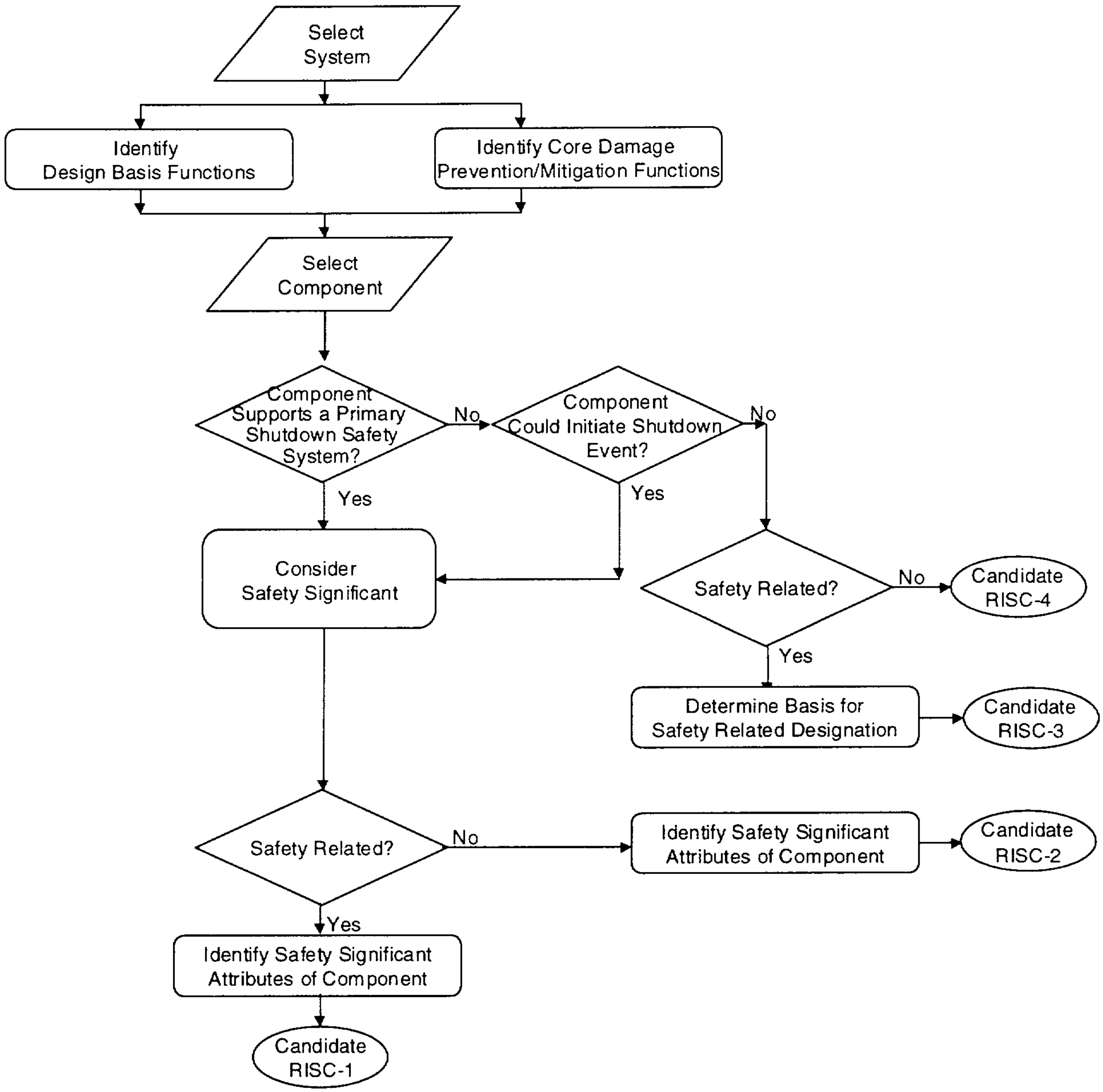
In this assessment, a primary shutdown safety system refers to a system, ~~which~~that has the following attributes:

- It has a ~~reasonable pedigree~~technical basis for its ability to perform the function.
- It has substantial margin to fulfill the safety function.
- It does not require extensive manual manipulation to fulfill its safety function.

If the risk importance process does not indicate that the component is safety significant, then the question of ~~safety related~~safety-related is asked. In the event it is a ~~safety related~~safety-related component, then the basis for that designation is questioned and the component is designated as a candidate for RISC-3. If the component is not ~~safety related~~safety-related, then it is a candidate for RISC-4.

[5] Each outage may be uniquely planned. However, the configuration control in place will maintain adequate safety and defense-in-depth. The Outage Risk Management Guidelines categorize the level of safety and specify the minimum acceptable number of systems for each safety function (e.g., sometimes referred to as the ORANGE condition).

Figure 3.4-2~~2.4-13~~
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS CREDITED IN NUMARC 91-06 PROGRAM



4 PREPARATION FOR IDP

In preparation for review by the integrated decision-making panel (IDP), the results and insights from the categorization process should be assembled into a form useful to the IDP and additional defense-in-depth information should be provided to assist the IDP in assigning the final categorization.

4.1 INTEGRAL ASSESSMENT

In order to provide the IDP with an overall assessment of the risk significance of SSCs, an integrated computation is performed using the available importance measures. This integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, seismic PRAs) by the fraction of the total core damage frequency contributed by that contributor. The following formulas define how such measures are to be computed for CDF. The same format can be used for LERF, if available.

Integrated Fussell-Vesely Importance

$$IFV_i = \frac{\sum_j (FV_{i,j} * CDF_j)}{\sum_j CDF_j}$$

Where,

IFV_i = Integrated Fussell-Vesely Importance of Component i over all CDF Contributors
$FV_{i,j}$ = Fussell-Vesely Importance of Component i for CDF Contributor j
CDF_j = CDF of Contributor j

Integrated Risk Achievement Worth Importance

$$IRAW_i = 1 + \frac{\sum_j (RAW_{i,j} - 1) * CDF_j}{\sum_j CDF_j}$$

Where,

$IRAW_i$ = Integrated Risk Achievement Worth of Component i over all CDF Contributors
$RAW_{i,j}$ = Risk Achievement Worth of Component i for CDF Contributor j
CDF_j = CDF of Contributor j

Integrated Risk Reduction Worth Importance

$$IRRW_i = 1 + \frac{\sum_j (RRW_{i,j} - 1) * CDF_j}{\sum_j CDF_j}$$

Where,

$IRRW_i$ = Integrated Risk Reduction Worth of Component i over all CDF Contributors
$RRW_{i,j}$ = Risk Reduction Worth of Component i for CDF Contributor j
CDF_j = CDF of Contributor j

Once calculated, an assessment should be made of these integrated values against the screening criteria of Fussell-Vesely >0.005 and RAW > 2. In no case will the integrated importance become higher than the maximum of the individual measures. However, it is possible that the integral value could be significantly less than the highest contributor, if that contributor is small relative to the total CDF/LERF.

4.2 DEFENSE IN DEPTH ASSESSMENT

In cases where the component is ~~safety related~~safety-related and found to be of low safety significance, it is appropriate to confirm that defense in depth is preserved. This discussion should include consideration of the events mitigated, the functions performed, the other systems that support those functions and the complement of other plant capabilities that can be relied upon to prevent core damage and large, early release.

Core Damage Defense-in-Depth

The initial ~~is~~ assessment should consider both the level of defense in depth in preventing core damage and to the frequency of the events being mitigated. Figure 4.2-1 below is an example of such an assessment~~:~~. This figure depicts the design basis events considered in the licensee's safety analysis report and considers the level of defense-in-depth available, based on the success criteria utilized in the PRA. This ensures that adequate defense-in-depth is available to mitigate design basis events.

Figure 4.2-1
Defense-in-Depth Matrix

Frequency	Design Basis Event	≥3 diverse trains OR 2 redundant systems	1 train + 1 system with redundancy	2 diverse trains	1 redundant automatic system
>1 per 1-10 yr	Reactor Trip Loss of Condenser				
1 per 10-10^2 yr	Loss of Offsite Power Total loss of Main FW Stuck open SRV (BWR) MSLB (outside cntmt) Loss of 1 SR AC Bus Loss of Instr/Cntrl Air			POTENTIALLY SAFETY SIGNIFICANT	
1 per 10^2-10^3yr	SGTR Stuck Open PORV/SV RCP Seal LOCA MFLB MSLB Inside Loss of 1 SR DC bus	LOW SAFETY SIGNIFICANCE CONFIRMED			
<1 per 10^3 yr	LOCAs Other Design Basis Accidents				

For example, if a PWR found that SSCs in the condensate system could be classified as low safety significant, this table could be used to qualitatively evaluate the safety significance. Since condensate is primarily relied upon as a secondary heat removal source following a reactor trip, the plant could confirm the low safety significance if three diverse trains or two redundant systems of heat removal are available. Many plants have three diverse trains of alternate feedwater makeup (e.g., turbine driven AFW, motor driven AFW and startup feedwater or diesel driven AFW) and many PWRs can utilize primary system bleed and feed as a means of heat removal. In these cases, the classification of condensate components as a low safety significant could be confirmed. If less defense in depth is available, that information should be provided to the IDP for their consideration in the final classification.

Containment Defense-in-Depth

Defense in depth should also be assessed for SSCs that play a role in preventing large, early releases. Level 2 PRAs have identified the several containment challenges that are important to LERF. These include containment bypass events such as ISLOCA (BWR and PWR) and SGTR (PWR), containment isolation failures (BWR and PWR), and early hydrogen burns (ice condensers and Mark III),

Before making the final decision on whether a SSC is categorized as low safety significance, the IDP should be provided with information on containment performance using the ~~consider the~~ following criteria:

Containment Bypass

- Can the SSC initiate or isolate an ISLOCA event?~~.~~
- Can the SSC isolate a faulted steam generator following a steam generator tube rupture event?~~.~~

Containment Isolation

- Does the SSC support containment isolation for containment penetrations that are:
  - >2" in diameter,
  - part of a system that is not considered closed as defined in GDC 57,
  - not normally closed or locked closed, and
  - not a part of a normally liquid filled system?~~.~~

Early Hydrogen Burns

- Does the SSC support operation of hydrogen igniters in ice condenser and Mark III containments?~~.~~

In cases where the answer to any of the above questions is "yes," the IDP should be informed that the SSC is potentially safety significant. If all of the above questions are answered "no," then low safety significance is confirmed.

In cases where SSCs are identified as safety significant, the safety significant attributes should be defined by the analyst familiar with the PRA. This involves identifying the performance aspects and failure modes of the SSC, that contribute to it being safety significant. These attributes are to be provided to the IDP~~ as input to the definition of treatment for RISC-1 and RISC-2 SSCs~~.

4.3 PRESENTATION OF RISK INFORMATION

The results of the compilation of risk information and safety significant attributes should be documented for the IDP's use. Figure 4.3-1~~2.4-12~~ provides an example, conceptual layout of the information that is generated by this process and could be useful for the IDP. This format is for the purposes of identifying what could be communicated and is not required.

At a minimum, the IDP should be provided with the following information for each SSC:

- Current classification of the SSC.
- The design basis function(s) supported by the SSC (for ~~safety related~~safety-related SSCs).
- The important to safety function(s) supported by the SSC (for important to safety SSCs).
- The PRA function(s) supported by the SSC.
- The results of the risk significance assessment for each hazard, and the integral assessment.
- Any applicable insights from sensitivity studies.
- The results of the defense-in-depth assessment.
- A summary of the basis for the classification recommendation to the IDP.
- A list of safety significant attributes for candidate RISC-1 and -2 SSCs.

In addition, it may be useful to have performed a preliminary sensitivity study as described in Section 2.4.4 in order to provide the IDP with insights regarding the potential cumulative impacts of changes in treatment.

The assessment of overall safety significance from the PRA involves consideration of the results of the categorization for each individual hazard and the integral assessment. The following guidelines are provided to assist in the communication of the categorization results to the IDP:

- If the SSC was found to be safety significant based on the internal events PRA without consideration of sensitivity studies, then it should be recommended to the IDP as safety significant.
- If the SSC was found to be of low safety significant based on the internal events PRA, but was found to be potentially safety significant based on the fire, seismic, other external hazards, or shutdown PRA assessments, then the integral assessment should be relied upon.
- If the SSC was found to be safety significant based on sensitivity studies, this should be communicated to the IDP, along with the base and integral significance for each hazard.

Figure 4.3-1
EXAMPLE RISK-INFORMED SSC ASSESSMENT WORKSHEET

SSC(S) EVALUATED: ______________________

SAFETY-RELATED: YES { NO {

DESIGN BASIS FUNCTION(S) SUPPORTED: ______________________

PRA FUNCTIONS SUPPORTED: ______________________

		Potentially Risk Significant	Potentially Non-Risk Significant	Not Assessed	Comments
Internal Events	CDF				
	LERF				
Fire	CDF				
	LERF				
Seismic	CDF				
	LERF				
External Hazards	CDF				
	LERF				
Low Power/ Shutdown	CDF				
	LERF				
Integral Assessment	CDF				
	LERF				

SENSITIVITY STUDY RESULTS:

DEFENSE-IN-DEPTH/COMMON CAUSE ASSESSMENT:

BASIS FOR RECOMMENDED SSC CLASSIFICATION:

SAFETY SIGNIFICANT ATTRIBUTES: ______________________

4.4 EVALUATION OF RECOMMENDED CHANGES

The final step in the process of categorizing SSCs into risk-informed safety classifications involves the evaluation of the risk implications of changes in special treatment. This process involves three primary components:

- Define Treatment Changes
- Conduct Sensitivity Studies of Potential Risk Implications
- Define Performance Monitoring Program

In general, since one of the guiding principles of this process is that changes in treatment should not degrade performance for RISC-3 SSCs, and RISC-2 SSCs would be expected to maintain or improve in performance, it is anticipated that there would be little, if any, net increase in risk.

The first step in performing this assessment involves the identification of the specific changes in treatment of SSCs that may impact performance. This qualitative assessment should consider the specific treatment identified in the licensees programs and the performance monitoring established.

The second step is to perform sensitivity studies using the available PRAs to evaluate the potential impact on CDF and LERF. This step is useful because the importance measures used in the initial safety significance assessment were based on the individual SSCs considered. Changes in performance can influence not only the importance measures for the SSCs, ~~which~~ that have changes in performance, but also others. Thus, the aggregate impact of the changes should be evaluated to assess whether new risk insights are revealed. Sensitivity studies should be realistic. For example, increasing the unreliability of all RISC-3 SSCs by a factor of 2 to 5 could ~~represent a bounding impact on SSC performance.~~ provide an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of all low safety significant SSCs. Both the random and common cause failure events should be increased for failure modes expected to be impacted by the changes in special treatment. The factor of 2 to 5 is appropriate as a sensitivity because it is representative of the change in reliability between a mean value and an upper bound (95th percentile) for typical equipment reliability distributions. For example, for a lognormal distribution the ratio of 95th percentile to mean value would be approximately 2.4 for a error factor of 3 and 3.5 for an error factor of 10.

~~Likewise, r~~Reducing the unreliability of RISC-1 and RISC-2 SSCs by a similar factor may be called for, depending upon the specific changes in special treatment. The changes in CDF and LERF computed in such sensitivity studies should be compared to the risk acceptance guidelines of Reg. Guide 1.174 as a measure of their acceptability. In addition, importance measures from these sensitivity studies can provide insight as to which SSCs and which failure modes are most significant.

It is noted that the recommended FV and RAW threshold values used in the screening may be changed by the PRA team following this sensitivity study. If the risk evaluation shows that the changes in CDF and LERF as a result of changes in special treatment requirements are not within the acceptance guidelines of the Regulatory Guide 1.174, then a lower FV threshold value may be needed (e.g., 0.001) for a re-evaluation of SSCs risk ranking. This may result in re-classifying some of the candidate low safety significant SSCs as safety significant SSCs.

The third step of the overall risk evaluation is to review the performance monitoring ~~called for by the IDP~~ in conjunction with the results of the risk sensitivity studies to determine the monitoring strategies. This process should compare the assumptions of the risk sensitivity studies, the results of the sensitivity studies and the monitoring strategies to determine whether additional monitoring is called for in order to maintain risk within an acceptable regime. For example, if the sensitivity studies indicate that, even with bounding SSC performance assumptions, the risk will remain within acceptance guidelines, and the bounding performance assumptions are supported by monitoring programs, then no changes would be necessary. If, however, the risk sensitivity studies identified that changes in the performance of specific SSCs could cause the computed risks to exceed the acceptance guidelines, then additional monitoring may be called for.

The results of this sensitivity study should be provided to the IDP as an indication of the potential aggregate risk impacts. These sensitivity studies should be re-visited when the IDP has completed its final categorization to assure that the conclusions regarding the potential aggregate impact have not changed significantly. If the categorization of SSCs is done at different times, the sensitivity study should consider the potential cumulative impact of all SSCs categorized, not individual systems or components.

5 INTEGRATED DECISION-MAKING PANEL REVIEW & CLASSIFICATION

The Integrated Decision-making Panel (IDP) uses the information and insights compiled in the initial categorization process and combines that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of SSCs.

5.1 PANEL MAKEUP & TRAINING

The IDP is composed of knowledgeable plant personnel whose expertise represents the important process and functional elements of the plant organization, such as operations, design and engineering (e.g., systems, electrical, I&C including information technology, nuclear), industry operating experience, and maintenance. The panel can call upon additional plant personnel or external consultants, as necessary, to assist in the resolution of issues.

The precise makeup of the panel is up to the licensee. Personnel availability to attend the majority, if not all meetings, is an important element in the selection of IDP permanent members. In general, there should be at least five experts designated as members of the IDP with joint expertise in the following fields:

- Plant Operations (SRO qualified),
- Design Engineering (including safety analyses),
- Systems Engineering,
- Licensing,
- Probabilistic Risk Assessment.

Members may be experts in more than one field; however, excessive reliance on any one member's judgement should be avoided.

The licensee should establish and document specific requirements for ensuing adequate expertise levels of IDP members, and ensure that expertise levels are maintained. Two key areas of expertise to be emphasized are experience at the specific plant being evaluated and experience with the plant specific PRA analyses relied upon in the categorization process.

The IDP should be aware of the limitations of the plant specific PRA and, where necessary, should receive training on the plant specific PRA, its assumptions, and limitations.

The IDP should be trained in the specific technical aspects and requirements related to the categorization process. Training should address, at a minimum:

- The purpose of the categorization,
- The risk-informed defense-in-depth philosophy and criteria to maintain this philosophy,
- PRA fundamentals,
- Details of the relied upon plant-specific PRA analyses, including the modeling scope and assumptions,
- The role of risk importance measures including the use of sensitivity studies, and
- The assessment of SSC failure modes and effects.

Each of these topics should be covered to the extent necessary to provide the IDP with a level of knowledge sufficient to evaluate and approve SSC categorization using both probabilistic and deterministic information.

IDP decision criteria for categorizing SSCs as safety significant or low safety significant will be documented. A consensus process should be used for decision-making. Differing opinions shall be documented and resolved.

The IDP should perform their activities in accordance with a procedure for determining the safety-significance of a SSC, and for the review of safety-significant functions and attributes to ensure consistency in the decision making process. The integrated decision process should, where possible, apply objective decision criteria and minimize subjectivity. The decisions of the IDP, including the basis, should be documented and retained as quality records for the life of the facility.

The IDP should be described in a formal plant procedure that includes:

- The designated chairman, panel members, and panel alternates;
- Required training and qualifications for the chairman, members, and alternates;
- Requirements for a quorum, attendance records, agendas, and meeting minutes;
- The decision-making process;
- Documentation and resolution of differing opinions; and
- Implementation of feedback/corrective actions.

5.2 IDP PROCESS

The preliminary classification information generated as part of the categorization process, including consideration of the role each SSC plays in the plant-specific risk analyses and defense-in-depth, is provided to the IDP for review and final classification. The overall classification process to be used by IDP is shown in Figure 5.1-1~~2.4-13~~.

As part of the initial categorization effort, SSCs ~~which~~that have similar functions and similar roles in the plant PRA analyses are identified and preliminarily categorized as RISC-1, -2, -3, -4. The IDP could review this preliminary categorization either by individual SSC or by groups of SSCs. In some cases, where the functional role of multiple SSCs is similar, it may be useful to consider those SSCs at the same time. For example, the suction and discharge isolation valves on a pump, may have similar performance and functional impacts and could be considered together. The initial steps of the IDP involve review of the primary technical bases for the initial categorization: the SSC function(s) and the basis for the categorization. The purpose of this review is for the IDP to determine, based on its composite knowledge of the plant, whether the SSC has been appropriately reflected in the categorization. The appropriateness of the manner in which the SSC has been reflected should be judged based on the scope of functions considered and the manner in which the PRA analyses incorporate those functions. If the IDP determines that the SSC has not been appropriately reflected, then it is re-evaluated based on the insights from the IDP.

Review of Safety Significant SSCs (RISC-1 & -2)

For those SSCs determined to be appropriately reflected in the categorization, the IDP will evaluate the key aspects of the recommended categorization. For RISC-1 and RISC-2 SSCs, if the IDP has determined that the SSC was appropriately reflected and it was categorized as RISC-1 or RISC-2, then the IDP can not move that SSC to a less safety significant category. For RISC-1 SSCs, the IDP reviews the SSC attributes identified in the categorization process including ~~both~~ the design basis attributes (for RISC-1), any important to safety attributes (for RISC-2) and any additional attributes ~~which~~that were identified as important to the core damage prevention and mitigation functions of the SSC ~~outside the design basis~~. ~~For RISC-2 components, the IDP review will focus on attributes which were identified as important to the core damage prevention and mitigation functions of the SSC since these SSCs have no safety design basis.~~

SSCs which have high failure probabilities (usually indicative of screening values) and meet the screening criteria solely on the basis of Fussell-Vesely importance, may have been identified as candidate safety significant. The reasons for these SSCs being classified as candidate safety significant should be reviewed by the IDP to determine whether special treatment will have any impact on the ability of these SSCs to perform their function. In many cases, special treatment will have little or no impact on such SSCs. If the IDP determines that this is the case, it may decide to classify the SSC as low safety significant.

Review of RISC-3 SSCs

The SSCs initially categorized as RISC-3 are safety-related SSCs ~~which~~that were found in the categorization process to be of low safety significance. The IDP's role for these SSCs is to perform a complete risk-informed assessment of the SSC categorization including consideration of the risk information, defense-in-depth and safety margins.

Review of Risk Information

For SSCs that have not been identified as safety significant by the safety significance process in Section 2.4.2, the IDP should review the results to determine whether these SSCs are not implicitly depended upon in the PRA. The IDP determines if:

- Failure of the SSC will significantly increase the frequency of an initiating event, including those initiating events originally screened out of the PRA based on anticipated low frequency of occurrence.

- Failure of the SSC will fail a safety function, including SSCs that are assumed to be inherently reliable in the PRA (e.g., piping and tanks) and those that may not be explicitly modeled (e.g., room cooling systems, and instrumentation and control systems).

- The SSC is necessary for safety significant operator actions credited in the PRA, including instrumentation and other equipment called for in procedures.

- Failure of the SSC will result in failure of safety significant SSCs in a manner ~~which~~that poses a risk impact (e.g., through spatial interactions).

If any of the above conditions are true, the IDP should use an evaluation to determine the impact of relaxing requirements on SSC reliability and performance.

Review Defense-In-Depth Implications

When categorizing SSCs as low safety significant, the IDP should consider whether the defense-in-depth philosophy is maintained. Defense-in-depth is considered adequate if the overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure that no significant increase in risk will occur by the change in special treatment, and that:

- Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release (Section 4.2);
- System redundancy, independence, and diversity is preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters (Section 4.2);
- There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design; and
- Potential for common cause failures is taken into account in the risk analysis categorization.

If any of the above conditions are not true, the IDP should perform a qualitative evaluation to determine the impact of relaxing requirements on SSC reliability and performance. Low safety significance can still be assigned, if one or more of the following are true:

- Relaxing the requirements will have minimal impact on the failure rate increase.
- Historical data show that these failure modes are unlikely to occur.
- Such failure modes can be detected in a timely fashion.

SSCs identified as low safety significant in the initial categorization process, but having potential safety significance if common cause failure is assumed, should be reviewed by the IDP to determine appropriate strategies for reducing the potential for common cause failures and strategies for detection of failures. This could include recommending staggered testing, inspection and/or calibration of equipment.

Review Safety Margin Implications

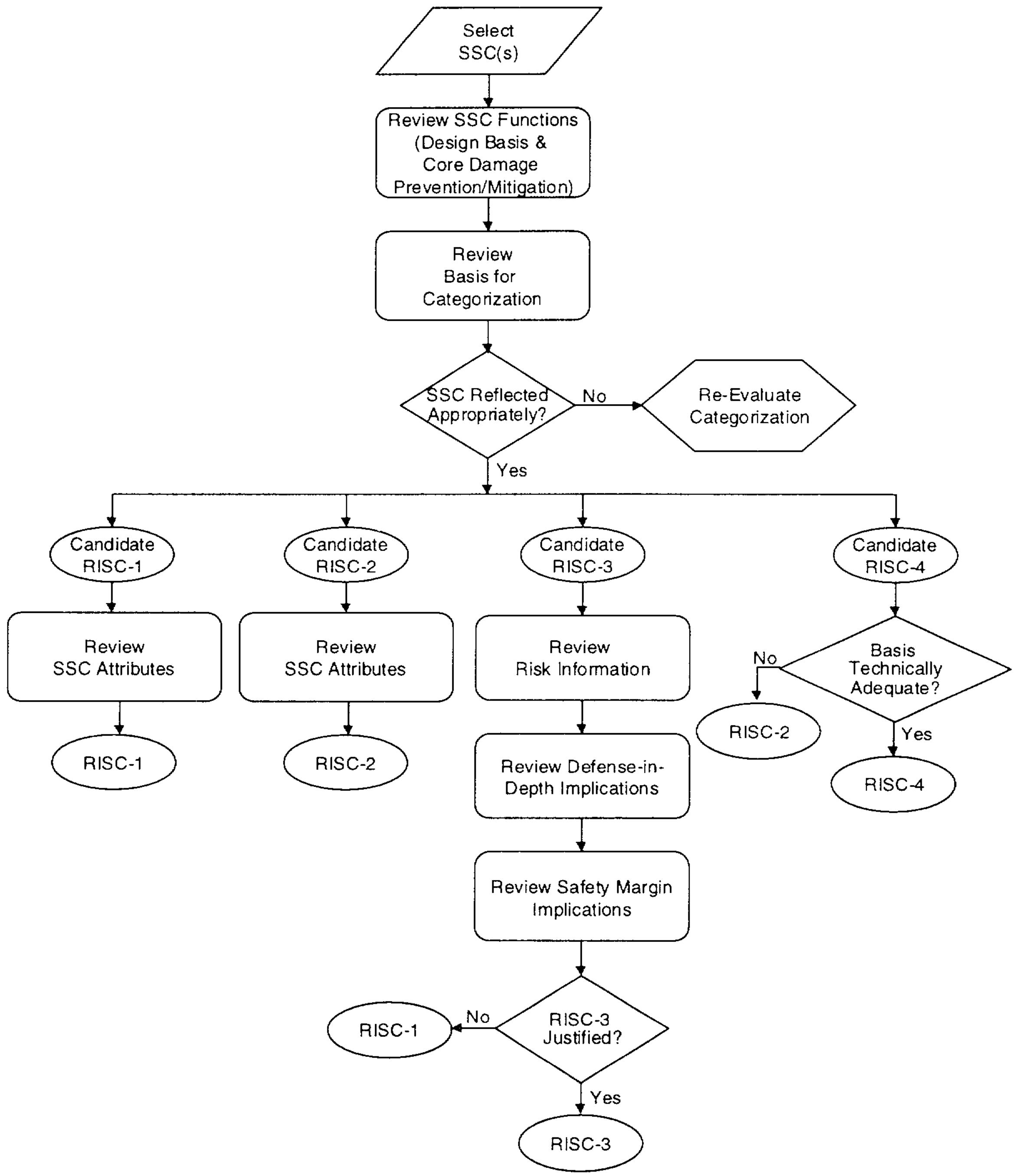
The treatment of low safety significant SSCs maintains design basis functions. Therefore, the functional performance of these SSCs will be assured and safety margin will be unaffected. The potential reliability impacts of the treatment changes are assessed in the sensitivity study to assure that potential changes in CDF and LERF are not significant. Consequently, no specific assessment of safety margin is required by the IDP. However, the IDP should qualitatively review each SSC classified as RISC-3 to ensure that no significant impacts on safety margin would be expected.
~~When categorizing SSCs as low safety significant, the IDP shall demonstrate that there is a sufficient safety margin to account for uncertainty in the engineering analysis and in the supporting data. Safety margin shall be incorporated when determining performance characteristics and parameters (e.g., component, system, and plant capability) or when defining mission success criteria (e.g., the number of system trains required to mitigate an initiating event or the ability of an SSC to perform in a certain environment). The amount of margin should depend on the uncertainty associated with the performance parameters in question, the availability of alternatives to compensate for adverse performance, and the consequences of failure to meet the performance goals. Demonstration of available safety margins shall be accomplished by use of data from plant operations or research studies, or by use of analyses using established engineering codes and standards or NRC-approved alternatives.~~

~~Upon completion of the review of the risk information, defense in depth, and safety margins, the IDP must come to a judgement that the categorization of the SSC as low safety significant is justified. If such a judgement can not be made, then the IDP can re-categorize the SSC to RISC-1. In doing so, however, the attributes of the SSC will have to be identified to ensure that any core damage prevention and mitigation attributes, which the IDP felt were significant, are included in future treatment.~~

Review of RISC-4 SSCs

The SSCs initially categorized as RISC-4 are non-safety-related SSCs ~~which~~that were found in the categorization process to be of low safety significance. The IDP's role for these SSCs is to ensure that the basis used in the categorization is technically adequate. For SSCs which are important to safety, the IDP must consider if the risk information used in the categorization process provides an adequate basis for categorizing the SSC as RISC-4. In general, the risk analyses should address the SSC function(s) which caused it to be originally classified as important to safety in order for a RISC-4 classification to be justified. If the IDP concludes that the categorization of the SSC as low safety significant is not justified, then the IDP can re-categorize the SSC to RISC-2. In doing so, however, the attributes of the SSC will have to be identified to ensure that any core damage prevention and mitigation attributes, ~~which~~that the IDP felt were significant, are included in future treatment.

Figure 5.1-1~~2.~~ ~~4-15~~
IDP PROCESS



5.3~~2.4.~~4 EVALUATION OF RECOMMENDED CHANGES

The final step in the process of categorizing SSCs into risk-informed safety classifications involves the evaluation of the risk implications of changes in special treatment. This process involves three primary components:

- Define Treatment Changes
- Conduct Sensitivity Studies of Potential Risk Implications
- Define Performance Monitoring Program

In general, since one of the guiding principles of this process is that changes in treatment should not degrade performance for RISC-3 SSCs, and RISC-2 SSCs would be expected to maintain or improve in performance, it is anticipated that there would be little, if any, net increase in risk.

The first step in performing this assessment involves the identification of the specific changes in treatment of SSCs that may impact performance by the licensee special treatment program owner. This qualitative assessment should consider the specific treatment identified in the licensees programs and the performance monitoring established.

The second step is to perform sensitivity studies using the available PRAs to evaluate the potential impact on CDF and LERF. This step is useful because the importance measures used in the initial safety significance assessment were based on the individual SSCs considered. Changes in performance can influence not only the importance measures for the SSCs~~, which~~that have changes in performance, but also others. Thus, the aggregate impact of the changes should be evaluated to assess whether new risk insights are revealed. Sensitivity studies should be realistic. For example, increasing the unreliability of all RISC-3 SSCs by a factor of 2 to 5 could ~~represent a bounding~~ provide an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of all low safety significant SSCs~~impact on SSC performance~~. Both the random and common cause failure events should be increased for failure modes expected to be impacted by the changes in special treatment. The factor of 2 to 5 is appropriate as a sensitivity because it is representative of the change in reliability between a mean value and an upper bound (95^{th} percentile) for typical equipment reliability distributions. For example, for a lognormal distribution the ratio of 95^{th} percentile to mean value would be approximately 2.4 for a error factor of 3 and 3.5 for an error factor of 10. ~~Likewise, r~~

Reducing the unreliability of RISC-1 and RISC-2 SSCs by a similar factor may be called for, depending upon the specific changes in special treatment. The changes in CDF and LERF computed in such sensitivity studies should be compared to the risk acceptance guidelines of Reg. Guide 1.174 as a measure of their acceptability. In addition, importance measures from these sensitivity studies can provide insight as to which SSCs and which failure modes are most significant.

It is noted that the recommended FV and RAW threshold values used in the screening may be changed by the PRA team following this sensitivity study. If the risk evaluation shows that the changes in CDF and LERF as a result of changes in special treatment requirements are not within the acceptance guidelines of the Regulatory Guide 1.174, then a lower FV threshold value may be needed (e.g., 0.001) for a re-evaluation of SSCs risk ranking. This may result in re-classifying some of the candidate low safety significant SSCs as safety significant SSCs.

The third step of the overall risk evaluation is to review the performance monitoring called for by the IDP in conjunction with the results of the risk sensitivity studies to determine the monitoring strategies. This process should compare the assumptions of the risk sensitivity studies, the results of the sensitivity studies and the monitoring strategies to determine whether additional monitoring is called for in order to maintain risk within an acceptable regime. For example, if the sensitivity studies indicate that, even with bounding SSC performance assumptions, the risk will remain within acceptance guidelines, and the bounding performance assumptions are supported by monitoring programs, then no changes would be necessary. If, however, the risk sensitivity studies identified that changes in the performance of specific SSCs could cause the computed risks to exceed the acceptance guidelines, then additional monitoring may be called for.

6 ~~3~~ TREATMENT OF RISK-INFORMED SAFETY CLASS STRUCTURES, SYSTEMS AND COMPONENTS

This section addresses the application of controls and treatment specifications for each of the four RISC SSC categories consistent with the safety-significance.

Licensee personnel who are the special treatment program owners are responsible for making changes to the specific special treatment requirements for SSCs under review. ~~As necessary and appropriate, P~~Program owners should ~~may~~ call upon additional plant personnel (system, design or PRA engineers) or external consultants to assist in the resolution of issues and the decision making process on the application of appropriate treatment. Once the program has been amended for one or more systems that have been risk-informed, the program changes are reviewed by the plant oversight group established for the review and approval of equipment modifications, and changes in procedures and programs. (This task may be delegated to the IDP)

These changes would generally be expected to maintain or improve SSC performance. For RISC-3 SSCs, changes in SSC treatment would be expected to have minimal impact on SSC performance, and that there would be sufficient confidence that the design bases function would be satisfied. ~~Nevertheless, the licensee management (IDP) should review the changes for these SSCs to assess how SSC performance may be impacted.~~

It is not necessary to modify or change SSC treatment just based on the results of the risk-informed categorization. Before making the decision to adjust treatment requirements, a licensee should first review the existing controls, specifications and SSC performance history, if available. An assessment should be ~~is~~ made of whether the SSC's past performance or existing treatment provisions (e.g., procurement, engineering specifications, etc.,) provide reasonable assurance that the ~~safety-significant~~ design bases functions[6] or the safety-significant beyond design bases functional requirement(s) identified in the §50.69 categorization process~~risk-informed evaluation process, or the functions specifically required by regulation or the safety analyses required by regulation~~ will be satisfied. Based on the results of these evaluations, a licensee determines the need to adjust treatment controls consistent with the safety significance of the functional attribute under review.

NRC technical requirements and the design process for RISC-1, RISC-2 and RISC-3 SSCs are not changed through implementing Section 50.69. Also, implementing §50.69 does not change the design bases.~~engineering specifications.~~

[6] As used in this section, the term function or design function relates to the interpretation provided in NEI 96-07, *Guidelines for 10 CFR 50.59 Safety Evaluations*, i.e., design bases functions or design functions that directly support the design bases functions as defined in NEI 97-04, *Design Bases Program Guidelines*.

NRC technical requirements are assessed~~,~~ and ~~where necessary,~~ improved under a separate activity, *Risk-Informing NRC Technical Requirements (Option 3 to SECY 98-300).*

As used in this document, the term design bases relates to the 10 CFR 50.2 definition of design bases. The term "beyond design bases" relates to those functions that are not part of the §50.2 design bases, i.e., the design functions required by regulation. A system's design may be based on power production needs, but since it is available it may also be used to mitigate or prevent a design bases event. The system is not credited in the §50.2 design bases, and therefore the function for the power production component is considered "beyond the [§50.2] design bases." This is an example of where risk-informed, performance-based regulation identifies and emphasizes latent safety enhancements that are already part of the non-regulated design. The newly identified beyond design bases functions provide increased safety assurance, provide an increased awareness of safety-significant functions and will further improve the focus on safety. ~~Assuring beyond design bases functions are satisfied enhances plant safety.~~

> *Example*: the feedwater system is not credited with providing a safety-injection function, yet in some scenarios, which are not part of the §50.2 design bases, the feedwater system can prevent and mitigate core damage.

Based on the categorization a system may have safety-significant and low safety-significant components within that system, or a safety-significant component may have low safety-significant sub-components or piece-parts. In each case treatment is applied consistent with the safety significance of the item or system under consideration.

~~5.4~~*6.1* ~~3~~ TREATMENT OF RISK-INFORMED SAFETY CLASS 1 SSCS

Risk-Informed Safety Class 1 SSCs are safety-related SSCs that the §50.69 categorization ~~risk evaluation~~ process has categorized as safety significant.

In general, there is no change to the regulatory treatment for these safety-related, safety-significant SSCs.

In specific instances for a RISC-1 SSC, the §50.69 ~~risk-~~evaluation process may identify an additional or different safety-significant function that is a "beyond design bases" function. These additional safety-significant functions should be ~~are~~ documented~~, as appropriate.~~ in the design bases documents and the design record files. In such cases, an engineering evaluation[7] should be performed to

[7] An evaluation is defined as an analysis (traditional or computer calculations), a review of test data, a qualitative engineering evaluation, or a review of operational experience, or any combination of these elements.

determine ~~ation is made on~~ whether the equipment could satisfy this new function. The licensee performs a review of the existing design and associated licensee activities against the §50.69 categorization process assumptions and conclusions. The review should include the following areas

(i) Design record files
(ii) Performance history;
(iii) Maintenance history;
(iv) Record of deficiencies;
(v) Existing work practices, procedures, and quality controls;
(vi) Material certification, tests or analyses;
(vii) Procurement history;
(viii) Engineering (including service conditions) and procurement specifications, and
(ix) Design record files and operating experience review information.

The engineering evaluation should determine whether new controls are necessary and, if necessary, establish new performance monitoring requirements. The evaluation should also determine whether equipment modifications are necessary or whether operating procedures need to be changed.

–If there is not reasonable assurance that the newly identified safety-significant function could be satisfied, a licensee has two choices:

(i) Take action to impose controls or implement changes to the facility that would result in the function being satisfied, or
(ii) Assess the impact on the SSC categorization process of not satisfying the function[8].

A more complete discussion of the change control processes for licensees that choose to adopt §50.69 is provided in Section 7.0.~~a licensee has two choices: determine the impact of not crediting the newly identified function, or take action to provide reasonable assurance that the newly identified safety function will be satisfied.~~

The identification and satisfaction of "beyond design bases" safety-significant functions enhances the current safety capabilities of the plant. These newly credited functions provide additional safety assurance above and beyond the current acceptable levels of safety. As such, it is appropriate and acceptable for industrial level (balance-of-plant) ~~commercial level~~ controls and practices to be applied to provide reasonable assurance that the "beyond design bases" functions

[8] While not crediting a beyond design bases function would not degrade plant safety below that defined by NRC requirements, its impact on the SSC categorization process needs to be assessed because it could change the categorization and treatment of high and low safety significant SSCs.

will be satisfied. A licensee should document the basis for determining that the SSC will satisfy the new safety-significant function.

A licensee's existing plant performance monitoring program, which includes the 10 CFR 50.65 performance assessment program and the existing corrective action program provide the necessary tools for assuring resolution of deficiencies. These programs also provide assurance ~~and continuing assurance~~ that the safety-significant functions will be restored if a degraded condition occurs~~satisfied~~. In addition, the periodic update of the PRA, which incorporates plant specific and industrywide ~~based on~~ operating experience ~~will~~ provides additional insights into the effectiveness of a licensee's categorization and corrective action programs for RISC-1 SSCs.

~~5.4.1.1~~Reporting Requirement for a Failure of a RISC-1 or RISC-2 Beyond Design Bases Function

Under §50.69, the current scope of §50.73, *License Event Reports,* is expanded to encompass safety-significant beyond design bases functions that have been identified by the §50.69 categorization process. A licensee event report that is consistent with the requirements in §50.73(b) is submitted to the NRC for an event or condition that alone prevented the satisfaction of a RISC-1 or RISC-2 safety-significant beyond design bases function. Events covered may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction deficiencies that would have prevented the beyond design bases function from being satisfied. Component failures need not be reported if redundant equipment was available to perform the required safety-significant function.

Other §50.73 and §50.72 reporting requirements continue to be applied to other RISC-1 and RISC-2 SSCs deficiencies as described in existing guidance.

~~Change Control Process for Beyond Design Bases Functions~~

~~RISC-1 SSCs are subject to §50.59. In addition, for RISC-1 SSCs that have a "beyond design bases" function, a licensee's configuration control program, which includes the §50.59 change control process, is adjusted to include a provision that provides reasonable assurance that RISC-1 safety-significant (including beyond design bases) function(s) will be satisfied following a facility change that involves a RISC-1 SSC. This additional change control provision would remain until the process is changed in response to the implementation of a risk-informed §50.59 process.~~

~~Where applicable, the additional change control provision determination is based on analyses (quantitative or qualitative) or on a combined quantitative and qualitative evaluation of the change and how it impacts the original design or operational bases. The information contained in the modification package, and if~~

~~necessary, the design record file, provides the detailed basis for the determination that the beyond design bases safety function will be satisfied. Each proposed change is supported by engineering information, that may include but is not limited to, drawings, specifications, narrative description, design evaluations, installation and testing requirements, associated procedure changes (if any), revised analyses (if any) and similar information. This information, often referred to as the change package, demonstrates the safety and effectiveness of the change and provides the basis for management approval of its implementation.~~

> ~~If the determination or licensee management concludes that there is insufficient assurance that the "beyond design basis" safety function would be satisfied following the implementation of a change, a licensee assesses the change against the minimal increase in risk standard defined in §50.59.~~

~~Following the completion of the evaluation, a licensee would follow the process described in the NEI 96-07, the implementing guidance for §50.59.~~
~~The engineering and operations documents associated with RISC-1 SSCs are already included in the scope of controlled documents for the plant. Information and action taken in response to the implementation of §50.69 relating to "beyond design bases" conditions should be documented in the engineering record files.~~

~~3.1.1 Example 1: BWR Containment Vent Valves (RISC-1)~~

~~Existing safety related functions include isolation of containment penetrations. The valves are required to close and remain closed under design basis conditions. In adopting §50.69, the risk-informed evaluation process categorized the vent valves as a safety significant (RISC-1 SSC) because, in addition to the containment isolation function, the valves need to open in specific emergency conditions to control containment pressure to prevent a catastrophic failure of containment. This is a "beyond design bases" function and provides an additional mitigation capability over and above that provided by the design bases. It enhances the protection of public health and safety.~~

~~An evaluation of existing engineering specifications, plant operations, design analyses, quality controls, and testing programs was performed to determine whether there was reasonable assurance that valves would open under the conditions requiring the venting of containment. The conclusion was that the existing design and controls provide reasonable assurance that the containment vent function will be satisfied. The plant's IST program was amended to include the opening function for these valves (stroke test). No other changes to controls for the valves, operators and the associated supporting equipment (electrical power supplies, air supply & I&C) were made.~~

~~The configuration control program, which includes the §50.59 process, was amended to include an evaluation of RISC-1 "beyond design bases" functions to~~

~~provide reasonable assurance that the safety-significant functions will be satisfied following a change that affects the valves.~~

~~The licensee documented its conclusions and its basis for the determination. The existing engineering records and controls (design, procurement, etc.....) already were included in the list of controlled documents and records for the plant.~~

~~5.4.23~~2Example ~~1~~2: PWR Pressurizer PORVs (RISC-1)

Existing safety-related functions include pressure-retaining boundary and opening to relieve pressure. In adopting §50.69, the risk-informed evaluation process categorized the PORVs as a safety-significant (RISC-1 SSC) because, in addition to the pressure retaining boundary, the valves can be credited to support "bleed and feed" heat removal capabilities, a "beyond design bases" function. The valve provides an additional method for mitigation and prevention over and above that for the designed safety-related function. When credited, it provides an enhancement to the protection of public health and safety.
NOTE: Given the availability of safety relief valves, the risk-evaluation process did not identify the pressure relief function as safety-significant. As a result, the PORV pressure relief function could be classified as RISC-3 and balance of plant (industrial) controls would apply. If so classified, then it would be necessary to retain the safety relief valves as RISC-1 for the pressure relief function.

There is no change in requirements or commitments associated with the pressure-retaining function.

The bleed and feed characteristic is not included in the [§50.2] design bases or credited in the safety analyses. An engineering evaluation and review of existing engineering specifications, plant operations, design analyses, quality controls, and testing programs was performed to determine whether the existing controls, including design and plant configuration, provide reasonable assurance that the "bleed and feed" function will be satisfied. The plant's IST program was amended to include a functional stroke test of this valve during a refueling outage. The configuration control process was amended to evaluate changes to provide reasonable assurance that the safety-significant functions will be satisfied following a change that affects the valves.

The configuration control program~~, which includes the §50.59 process,~~ was amended to include an evaluation of RISC-1 "beyond design bases" functions to provide reasonable assurance that the safety-significant functions will be satisfied following a change that affects the valves.

No other changes were made to controls for the valves and the associated supporting equipment (electrical power supplies & I&C).

The licensee documented its conclusions and its basis for the determination. The existing engineering records and controls (design, procurement, etc....) already were included in the list of controlled documents and records for the plant.

~~5.4.33.3~~Example 2~~3~~: Isolation Valves on the Suction Line of the Startup Auxiliary Feedwater Pump (RISC-1)

The existing safety-related function for these valves is to close and remain closed after a seismic event to perform this isolation and prevent draining of the ~~safety related~~safety-related water source. The §50.69 risk-informed evaluation process identified an additional safety-significant function. The startup auxiliary feedwater pump is an important source of feedwater following most reactor trips and the isolation valves must be open, and remain open to support the newly identified function.

NOTE: The two ~~safety related~~safety-related isolation valves are provided on the suction line of the startup auxiliary feedwater pump to isolate the seismic designed water source from the non-seismic startup pump. The startup auxiliary feedwater pump is a non-safety-related, non-seismic pump that uses the same water source as the ~~safety related~~safety-related auxiliary feedwater pumps.

An engineering evaluation was performed and determined that the valves would remain functional. Existing maintenance, operating and testing procedures, plus design and procurement specifications were evaluated. The valves were normally tested every quarter. The test procedure was expanded to include a test of the opening function at the same periodicity. ~~Future activities (procurement, maintenance, modifications, etc.) on these valves would be performed in accordance with the current RISC-1 requirements.~~Future activities (procurement, maintenance, modifications, etc.) on these valves would be performed in accordance with balance of plant (industrial) practices because startup feedwater is a "beyond design basis" function.

The configuration control program, which includes the §50.59 process, was amended to include an evaluation of RISC-1 "beyond design bases" functions to provide reasonable assurance that the safety-significant functions will be satisfied following a change that affects the valves.

The licensee documents its conclusions and its basis for the determination. The existing engineering records and controls (design, procurement, etc....) already were included in the list of controlled documents and records for the plant.

~~5.5~~6.2 ~~3~~ TREATMENT OF RISK-INFORMED SAFETY CLASS 2 SSCS

RISC–2 SSCs are nonsafety-related SSCs that a Section 50.69 risk-informed categorization~~evaluation~~ process has determined to be safety-significant

The identification and satisfaction of ~~the~~ RISC-2 safety-significant functions enhances the current safety capabilities of the plant. These newly credited functions provide additional safety assurance beyond the current acceptable levels of safety. As such, it is appropriate and acceptable for ~~commercial~~industrial level controls and practices to be applied to provide reasonable assurance that the safety-significant functions will be satisfied. The basis for determining that a RISC-2 SSC will satisfy a newly identified safety-significant function is documented in an engineering evaluation that is consistent with the station's procedures for balance-of-plant or important-to-safety SSCs. If a licensee determines that there is not reasonable assurance that the safety-significant function could be satisfied, a licensee has two choices:

(i) Take action to impose controls or implement changes to the facility that would result in the function being satisfied, or
(ii) Assess the impact on the SSC categorization process of not satisfying the function[9] ~~determine the impact of not crediting the newly identified function, or take action to provide reasonable assurance that the function will be satisfied.~~

A more complete discussion of the change control processes for licensees that choose to adopt §50.69 is provided in Section 7.0

For a majority of licensees that implemented 10 CFR 50.65 based on functional failures as opposed to maintenance preventable functional failures, the only changes associated with the programs for RISC-2 SSCs are linked to a licensee's configuration control and NRC 10 CFR 50.73 reporting programs[10]. With the exception of these two areas, the same regulatory requirements (§50.49, §50.63, §50.65 etc.,) and associated commitments are applied to RISC-2 SSCs to the same extent ~~to RISC-2 SSCs~~ as prior to the implementation of §50.69~~the risk-informed categorization process~~.

For RISC-2 SSCs, the ~~a~~ existing §50.65 performance monitoring program plus existing (~~commercial~~industrial (BOP) and, as applicable, augment quality) controls and specifications are sufficient. ~~For a majority of licensees, the monitoring program established by the maintenance rule is sufficient for assuring that the safety-significant functions will be satisfied providing the maintenance rule performance criteria were based on functional failures, and not just maintenance preventable functional failures.~~

Establishment of New Performance Criteria

[9] While not crediting a beyond design bases function would not degrade plant safety below that defined by NRC requirements, its impact on the SSC categorization process needs to be assessed because it could change the categorization and treatment of high and low safety significant SSCs.

[10] See Reporting Requirement for a Failure of a RISC-1 or RISC-2 Beyond Design Bases Function in Section 6.1

If a licensee's maintenance rule performance criteria were not established based on functional failures, e.g., only based ~~only using maintenance~~on maintenance-preventable functional failures (MPFFs), a licensee should review and~~, where appropriate,~~ establish new performance thresholds for RISC-2 SSCs. For a number of licensees, the existing performance criteria may be sufficient. The determination on the need for adjusting the performance criteria should be based on a ~~A~~ review ~~should be performed to establish~~ of the performance history record and, if available, an evaluation of the -existing licensee controls for the SSCs under review. ~~to assure the correct controls are in place. As applicable,~~ Such evaluations should include a review ~~the review may include~~ of the following areas:

(i) §50.69 categorization assumptions and results~~PRA assumptions and conclusions~~;
(ii) Performance history;
(iii) Maintenance history;
(iv) Record of deficiencies;
(v) Existing work practices, procedures, and quality controls;
(vi) Material certification, tests or analyses;
(vii) Procurement history;
(viii) Engineering (including service conditions) and procurement specifications~~.~~; and
(ix) Design record files and operating experience review information.

NOTE: For many licensees, the review of the safety-significant functions identified ~~PRA~~ by the SSC categorization process and the assumptions in the SSC categorization determination ~~functions and assumptions~~ should provide sufficient information. The performance of safety-significant functions by nonsafety-related SSCs to prevent or mitigate conditions, which are "beyond design bases" events, is included in PRAs based on various justifications. In performing the PRA, t~~T~~he availability of a nonsafety-related SSC to potentially perform a "beyond design bases" safety-significant function is based on consultation with design or systems engineering groups. ~~Generally, if engineering judgement is used, the PRA uses success criteria on SSC performance to evaluate whether the SSC can provide the function under the service conditions required. Uncertainty in this judgment is compensated by assigning a probability of failure to the SSC performance.~~ In general, where ~~In general,~~ engineering judgment is used it is based on operating history or knowledge of SSC or similar equipment performance under conditions, which may approximate to those represented by the PRA; e.g., a previous unusual event involving water passage through valves whose usual service conditions involve steam flow. Generally, the PRA uses success criteria on SSC performance to evaluate whether the SSC can provide the function under the service conditions required. Assigning a probability of failure to the SSC performance compensates uncertainty in this judgment. -External data sources, such as IDCOR, NRC research publications, Licensee Event Reports

(LERs), and the Institute of Nuclear Power Operations (INPO) reports may be consulted to determine if there is a precedent showing the SSC can perform as needed. Although these techniques are not as rigorous as traditional engineering specifications and testing, they provide reasonable assurance for the low probability service conditions being considered.

Based on the above information, a licensee can determine the optimum and practical performance criteria that will provide reasonable assurance that the safety-significant functions defined in the risk-informed evaluation process are satisfied.

~~5.5.4~~Implementation of Additional RISC-2 SSC Controls

An indication of areas where controls may need to be enhanced is evaluated by comparing maintenance and performance information against the performance criteria. Changes in equipment controls should be effected through the application of the licensees root cause and corrective action programs to the areas of identified weakness. When completed, ~~Also,~~ this ~~such a~~ review ~~documents~~ documents the controls and specifications that provide reasonable assurance that the additional safety-significant functions will be satisfied.

During normal plant operations (power or shutdown, including refuelling), a~~A~~ licensee's monitoring and corrective action programs provide the necessary tools for assuring resolution of deficiencies and continuing assurance that the safety-significant functions will be satisfied. In addition, the update of the PRA based on operating experience will provide additional insights into the effectiveness of a licensee's categorization and corrective action programs for RISC-2 SSCs.

A licensee's ~~commercial~~industrial (BOP) controls are dispersed throughout the licensee's documentation; in department orders, procedures, and training programs. Appendix C~~A~~ to this guideline provides examples of the type of activities that should be included in ~~commercial~~industrial control programs. Repair and replacement activities would be governed by the original code of construction and engineering specifications. For some specific and unique SSCs that are subject to ASME requirements additional activities may be delineated in the code. (See ASME Section XI Code Case, *under development and schedule to be issued in 2001).*

Changes to controls and specifications for RISC-2 SSCs are documented. The design and operations documents for RISC-2 SSCs are added to the scope of controlled documents for the plant, if they are not part of the controlled documentation process. Information and action taken in response to the implementation of §50.69 relating to "beyond design bases" conditions should be documented in the engineering record files.

For RISC-2 SSCs that are already governed by regulations, such as, 10 CFR 50.49, *Environmental qualification of electric equipment important to safety for nuclear power plants;* 10 CFR 50.62, *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants;* and 10 CFR 50.63, *Loss of all alternating current power;* the existing controls, sometimes known as "augment quality controls", defined in regulatory guidance documents, such as Regulatory Guide 1.155, would continue to be applied provided these regulations are applicable to the function resulting inclassification of the component as RISC-2.

In unique and specific instances, the §50.69 SSC categorization process may identify ~~For~~ ~~new~~new safety-significant beyond design bases seismic or other environmental attributes. ~~identified by the §50.69 risk-informed evaluation process.~~ In these cases, a licensee would evaluate the ability of the SSC to satisfy the newly identified functions ~~identified safety function (including the service condition requirements)~~ using an ~~commercial~~industrial standard of assurance. Appendices D and E ~~, i.e., standard balance-of-plant criteria.~~ describe such processes that are based on industry consensus standards. ~~The process would be based on an~~ engineering specifications and balance-of-plant evaluations. ~~review~~ Such evaluations provide reasonable (industrial level of assurance) ~~to determine~~ that the SSC would operate satisfactorily under the specified environmental conditions. An industrial level of assurance is acceptable because these newly identified functions provide an enhanced level of safety, above and beyond the standard defined in the licensing bases. Vendor specifications or licensee evaluations should be sufficient, and generally testing would not be required. Appendices D and E provide additional information on acceptable industrial practices for seismic and environmental qualification treatment.

10 CFR Part 21 does not apply to RISC-2 SSCs because they are not "basic components." This is consistent with the existing Part 21 regime, where Part 21 is not applicable to nonsafety-related SSCs that are currently governed by the regulations, e.g. §50.49 and §50.65.

~~For RISC-2 SSCs that are associated with a "beyond design bases" function, a provision is added to a licensee's configuration control program, which includes the §50.59 change control process. This new provision requires an evaluation to reasonably assure that the newly identified safety-significant function (including the service condition) will be satisfied following a change to facility (equipment or procedures) that affect RISC-2 SSCs (See RISC-1 Section for additional details). This additional control provision is necessary because, apart from specific regulations that focus on nonsafety-related SSCs, such as, §50.62, and §50.49, the current §50.59 process focuses on design bases, not "beyond design bases" functions. This additional change control provision would remain until the process is changed in response to the implementation of a risk-informed §50.59 process.~~

~~NRC §50.73, *Licensee event report system,* requirements are modified per §50.69. For a licensee adopting §50.69, a licensee shall submit a licensee event report consistent with the requirements in §50.73(b) for an event or condition that alone prevented the satisfaction of the §50.69 RISC-2 SSC safety-significant function Events covered may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. Component failures need not be reported if redundant equipment was available to perform the required safety function.~~

~~10 CFR Part 21 does not apply to RISC-2 SSCs because they are not "basic components." This is consistent with the existing Part 21 regime, where Part 21 is not applicable to nonsafety-related SSCs that are currently governed by the regulations, e.g. §50.49 and §50.65.~~

~~5.5.2~~3 Example 1: Alternate AC Gas Turbine Generators (GTGs) (RISC-2)

Prior to the implementation of 50.69 the GTGs were considered "important to safety", but were not categorized as safety-related. They were included in the scope of the maintenance rule. They were recognized as safety significant because of their role in the mitigation of station blackout events

The §50.69 risk-informed categorization process identified the following function as safety-significant: Start and load by local operator action within one hour of the start of a station blackout event (maintenance of vital auxiliaries). This is consistent with the existing design function for this system.

The §50.65 monitoring program established performance criteria based on all failures, not just maintenance preventable functional failures. No changes to performance monitoring were required.

An evaluation of the existing controls determined the quality assurance requirements of Regulatory Guide 1.155 (August 1988), Appendix A have been applied to this system. The current equipment performance demonstrates that the existing controls have been adequate to maintain the safety-significant function. No changes were made to the existing controls.

~~Alternate AC requirements are included in the scope of §50.59. No additional configuration control processes needed to be established.~~

The licensee documented its conclusions and its basis for the determination. The existing engineering records and controls (design, procurement, etc.…) already are included in the list of controlled documents and records for the plant.

~~5.5.33~~6~~.2.2~~ Example 2: Instrument Air System (RISC-2)

~~5.4.3~~Prior to the implementation of 50.69, the Instrument Air system was categorized as nonsafety-related and not "important to safety". The system was included in the scope of the Maintenance rule. The §50.69 risk-informed evaluation process identified the system as safety significant with trip initiation under specific conditions as a safety significant function that is not included in the design bases of the facility. The specific trip initiations cause a scram with complicated actions because of the complexities of plant operation with a loss of instrument air. ~~The risk-informed evaluation process identified a potential improvement to the safety profile of the plant that is beyond the design bases of the facility. The implementation of §50.69 enhanced the protection of public health and safety.~~

~~A~~A review was performed of the current performance monitoring for this system. The current performance ~~criteria monitored this system at the plant level and~~ criteria monitored this system at the plant level and were based on~~considered~~ all functional failures of the system. This monitoring was determined appropriate for the safety significant function identified during the risk-informed evaluation. The current system performance was reviewed and found to be acceptable, so no additional controls were imposed.

The configuration control process was review to ensure that ~~amended to include an evaluation of~~ changes made to the Instrument Air system would be evaluated to provide reasonable assurance that following a change, ~~s to~~ the system would continue to satisfy the performance criterion.

The conclusions and its basis for the determination were documented. The existing engineering records and controls (design, procurement, etc.,...) were added to the list of controlled documents and records for the plant.

~~5.5.5~~3.2.3 Example 3: BWR Feedwater Pumps (RISC-2)

In a BWR, prior to the implementation of §50.69, the feedwater pumps were categorized as nonsafety-related SSCs, yet were included in the scope of the maintenance rule. In adopting §50.69, the risk-informed evaluation process categorized the feedwater pumps as safety-significant (RISC-2 SSCs) because they can be used to prevent and mitigate potential core damage events in scenarios that are not included in the design bases. These pumps provide additional methods for mitigation over and above the designed safety systems. When credited, they provide an enhancement to the protection of public health and safety.

The risk-informed evaluation identified the following functions as safety significant:

- Pressure boundary, and
- Water injection into reactor pressure vessel.

The pumps are already included in the maintenance rule-monitoring program. However, the licensee established the performance criteria based on only maintenance preventable functional failures, not on safety-significant failures. As a result, the licensee developed new performance criteria and controls for §50.69 implementation that also encompass the performance monitoring criteria for the maintenance rule. The licensee reviewed the following documentation:

- ~~PRA assumptions~~Categorization assumptions and conclusions associated with the feedwater pumps;
- Performance history,;
- Maintenance history;
- Record of deficiencies;
- Existing work practices, procedures, and quality controls;
- Procurement history; and
- Engineering and procurement specifications.

Based on these reviews new performance criteria were established. No changes to the controls for these pumps were necessary to provide assurance that the safety significant functional requirements would be satisfied. The basis was that the performance credited in the PRA to inject water was the same as the performance of the pumps to satisfy their function during normal operation. The performance is confirmed during pre-operational startup testing and continuously during normal operation. Existing testing, monitoring and corrective action practices provide reasonable assurance that the injection credited in the PRA will be available.

The configuration control program~~, which includes the §50.59 process,~~ was amended to require an evaluation to reasonably assure that the "beyond design bases" functions of the will be satisfied.

The existing engineering records and procedural controls (vendor manuals, procurement specifications, maintenance schedules and procedures) already were included in the list of controlled documents for the plant.

~~5.5.6~~3.2.4 Example 4: PWR Nonsafety-Related 4kv AC Power Buses (RISC-2)

In a PWR, prior to the adoption of §50.69, several 4kv power buses were categorized as nonsafety-related, yet were included in the scope of §50.65. In adopting §50.69, the risk-informed evaluation process categorized these 4kv AC power buses as safety-significant RISC-2. The basis for this determination was that these power sources may be used in "beyond design bases" configurations to prevent and mitigate an accident by providing power to components that could be used as an alternative method to safely shutdown the plant (e.g., use of condensate pumps as an alternate injection path for "beyond design bases" events).

An evaluation of the electrical coordination and loading characteristics was performed in accordance with station procedures and determined that the 4kV buses would satisfy the safety-significant functions. These nonsafety-related 4kv buses are already included in the monitoring program for the maintenance rule. In view of the history in satisfying the maintenance rule performance criteria, no additional evaluations or controls were needed. -Both unavailability and reliability (in terms of safety functional failures) performance criteria are included in the §50.65 monitoring program. The licensee's maintenance rule performance criteria are based on all failures, not just on those related to maintenance preventable functional failures. No additional monitoring was needed to provide reasonable assurance that the safety function would be satisfied. An existing evaluation had concluded that~~the 4kV buses would satisfy the safety significant function~~. ~~In view of the history in satisfying the maintenance rule performance criteria, no additional evaluations or controls were needed.~~

Future modifications or repairs to these 4kv AC power buses would be performed under the existing documented procedural controls (~~commercial~~industrial/balance-of-plant controls and processes). The configuration control program, which includes the §50.59 process, was amended to require an evaluation to reasonably assure that the "beyond design bases" functions of the will be satisfied for a change to a RISC-2 SSC.

The existing engineering records and procedural controls (vendor manuals, procurement specifications, maintenance schedules and procedures) already were included in the list of controlled documents for the plant.

~~5.5.7~~3.2.5 Example 5: PWR Normal Chilled Water System (RISC-2)

In a PWR, prior to the adoption of §50.69, normal chilled water (NCW) system was categorized as nonsafety-related and was included in the scope of the maintenance rule. In adopting §50.69, the risk-informed evaluation process identified (IDP decision) the NCW as safety-significant RISC-2 because this system could fail safety-related components that rely on normal HVAC systems as an alternate to emergency HVAC systems for operability.
NOTE: The NCW system is modeled in the plant PRA, yet based solely on the PRA, the system would not be categorized as safety-significant (there are no safety-significant components associated with this system).

The NCW system is already included in the monitoring program for the maintenance rule. Both unavailability and reliability (in terms of safety functional failures) performance criteria are included in the §50.65 program. The maintenance rule performance criteria are based on all failures, not just on those related to maintenance preventable functional failures. No additional monitoring is needed to provide reasonable assurance that the safety function would be satisfied.

In view of the history in satisfying the maintenance rule performance criteria, no additional controls were needed.

Future modifications or repairs to the NCW would be performed under the existing documented procedural control (~~commercial~~industrial/balance-of-plant controls and processes). The configuration control program, which includes the §50.59 process, was amended to require an evaluation to reasonably assure that the "beyond design bases" functions of the will be satisfied for changes to RISC-2 SSCs.

The existing engineering records and procedural controls (vendor manuals, procurement specifications, maintenance schedules and procedures) already were included in the list of controlled documents for the plant.

~~5.5.8~~Example 6: BWR Containment Vent Valves (RISC-1

Existing safety-related functions include isolation of containment penetrations. The valves are required to close and remain closed under design basis conditions. In adopting §50.69, the risk-informed evaluation process categorized the vent valves as a safety-significant (RISC-1 SSC) because, in addition to the containment isolation function, the valves need to open in specific emergency conditions to control containment pressure to prevent a catastrophic failure of containment. This is a "beyond design bases" function and provides an additional mitigation capability over and above that provided by the design bases. It enhances the protection of public health and safety.

An evaluation of existing engineering specifications, plant operations, design analyses, quality controls, and testing programs was performed to determine whether there was reasonable assurance that valves would open under the conditions requiring the venting of containment. The conclusion was that the existing design and controls provide reasonable assurance that the containment vent function will be satisfied. The plant's IST program was amended to include the opening function for these valves (stroke test). No other changes to controls for the valves, operators and the associated supporting equipment (electrical power supplies, air supply & I&C) were made.

The configuration control program was amended to include an evaluation of RISC-1 "beyond design bases" functions to provide reasonable assurance that the safety-significant functions will be satisfied following a change that affects the valves.

The licensee documented its conclusions and its basis for the determination. The existing engineering records and controls (design, procurement, etc.,...) already were included in the list of controlled documents and records for the plant.

~~5.6~~6.3 ~~3.3~~ TREATMENT OF RISK-INFORMED SAFETY CLASS 3 SSCs

RISC-3 SSCs are safety-related SSCs that have been categorized as not being safety-significant under the risk-informed evaluation methodology and that are directly and specifically referenced in a regulation or in a licensee's safety analyses (e.g., FSAR Chapter 15 analyses) required by regulation.

Configuration Control for RISC-3 SSCs

RISC-3 SSCs are subject to the 10 CFR 50.59 change control process. If the [§50.2] design bases are changed under Option 3, *Risk-Informing NRC Technical Requirements*, and are no longer applicable to specific RISC-3 SSCs, then the full §50.59 evaluation would not be required.

Where appropriate and practical, RISC-3 SSC performance should be monitored against functional criteria (e.g., functional inspections, tests, or operational performance reviews) set to provide sufficient confidence that the design bases functions will be satisfied. Where monitoring is impractical or would not provide the assurance that the safety-significant function would be satisfied, existing industrial (BOP) controls and procedures, including the use of condition monitoring and engineering evaluations are used to provide sufficient confidence that the required function will be satisfied.

When adopting §50.69, a licensee makes the following licensing commitment: ~~These SSCs are the subject of the following licensing commitment that supercedes all previous RISC-3 SSC commitments.~~

- For RISC-3 SSCs, ~~The application of~~ an ~~commercial~~industrial level performance monitoring program or, where monitoring is not appropriate or practical, industrial ~~commercial~~ level controls are applied to provide sufficient confidence ~~reasonable assurance~~ that the design bases ~~SSC~~ functional requirements~~, that are directly and specifically referenced or described in a regulation, or in the assumptions and conclusions of the plant specific safety analyses required by regulation.~~ will be satisfied.

Existing NRC commitments for RISC-3 SSCs should be reviewed and may be changed through the application of NEI 99-04, Rev.1, *Guideline for Managing NRC Commitments*. NEI 99-04 has been revised to reflect the impact of the §50.69 SSC categorization scheme. Section 7.0 provides additional guidance on the configuration control processes for licensees that adopt §50.69.
~~This is consistent with NEI 99-04, Guidelines for Managing NRC Commitment Changes *(in process of being amended)* that permits a licensee to change commitments associated with low safety-significant SSCs (RISC-3 or RISC-4~~

~~SSCs). No other regulatory commitments are applicable to these SSCs because they have been categorized as not having safety significance.~~

RISC-3 SSCs whose failure would not result in a failure to satisfy a design bases function may be categorized to RISC-4 SSC through the application of §50.59. Yet, a full §50.59 evaluation may not be required.

Application of New RISC-3 SSC Functional Criteria

~~Where appropriate, RISC-3 SSC performance would be monitored against functional criteria set to provide reasonable assurance that the functions directly referenced in the regulations, or directly and specifically referenced in the safety analyses required by regulation will be satisfied. Such P~~Performance ~~criteria~~ criteria are determined by the licensee and are set at ~~would be at~~ the plant, system, train or component level. In many cases, the functional performance criteria could be a subset of the §50.65 performance monitoring criteria.

If new functionality criteria need to be established, the criteria sh~~w~~ould be set by first determining the specific design bases functional ~~regulatory functional~~ requirement ~~or safety analyses function~~. A review should then be performed of documents, such as: ~~is then performed of the following documentation:~~

- The applicable regulation(s) and associated regulatory guidance document(s),
- ~~PRA assumptions~~§50.69 SSC categorization assumptions and conclusions,
- Design record files
- Performance history;
- Record of deficiencies,

The basis for the functional performance criteria should be documented and becomes part of the system's records.
~~The licensee then establishes functional criteria for the SSC that when satisfied provide reasonable assurance that the functions required by the specific regulation, or the plant specific safety analyses required by regulation will be satisfied. A comparison of the SSCs performance history against the new functional criteria should be made.~~

Failures to satisfy the RISC-3 functional performance criteria are addressed and resolved through a licensee's corrective action program. The functional and conditioning monitoring programs plus ~~Monitoring and the licensee's~~a licensee's corrective action program provide the necessary tools for assuring and determining that ~~resolution of~~ deficiencies have been resolved.~~and continuing assurance that the functions required by the applicable regulation will be satisfied. In addition, the update of the PRA will provide additional insights into the effectiveness of the categorization and the licensee's corrective action program~~

~~for~~
~~RISC-3 SSCs.~~

Application of Controls for RISC-3 SSCs

RISC-3 SSCs are subject to a licensee's industrial balance-of-plant controls. These controls are applied to provide sufficient confidence that the design bases functions will be satisfied, as demonstrated through the satisfaction of the functional performance or condition monitoring criteria.

~~Where monitoring is inappropriate or impractical, existing commercial (BOP) controls and procedures are used to provide reasonable assurance that the required function will be satisfied.~~ A review of procurement specifications, existing work controls and practices, and design files should be performed to establish controls that will provide sufficient confidence that the design bases function(s) will be satisfied.~~is performed to determine that there is reasonable assurance that the SSC will operate satisfactorily in accordance with the engineering specifications described in the design record file.~~

Seismic and Environmental Qualification Considerations

~~As appropriate,~~ As determined by the design and the §50.69 categorization process, eenvironmental attributes, such as,~~e.g.,~~ water immersion, seismic, fire, or harsh environment, are included in procurement specifications for replacement parts. In such cases condition monitoring and inspection sh~~w~~ould be sufficient for issues such as, seismic two-over-one conditions for RISC-3 SSCs, where component anchorage would be inspected.

For seismic, an evaluation should be performed to provide an industrial (balance-of-plant) confidence that the equipment will operate in a manner to satisfy its design basis functional requirements. Under §50.69 and in a manner similar to the ASME code cases for low safety-significant SSCs, it is acceptable to apply established industrial level seismic practices and standards to RISC-3 SSCs. These nationally recognized standards and established practices include methods for assessing equipment functionality under seismic conditions, equipment qualification criteria, and for determining seismic design loads.

RISC-3 equipment should be seismically qualified through an engineering evaluation or through an engineering evaluation in conjunction with a national consensus standard, such as the International Building Code (IBC 2000). Technical procurement specifications should be based on design requirements, the application of industrial standards, and, if necessary, a technical procurement evaluation, which concludes that the design bases functions will be satisfied.

Recent nationally recognized consensus standards for structural design and construction have included state-of-the-art criteria for determining seismic design loads. To allow the industry to gain more experience in implementing these improved standards an interim hybrid approach (Part 100, Appendix A criterion for seismic loads/Consensus Standard for equipment evaluation) has been

developed for addressing seismic conditions. The approach is described in Appendix E.
~~In procuring components that have seismic requirements, reference should be made to commercial consensus standards that have been developed for commercial non-nuclear applications in seismic areas. If there is no appropriate consensus standard, an evaluation[11] would be performed to provide reasonable (commercial level) assurance that the SSC would be able to satisfy the required seismic functional criteria. Guidance for such evaluations is provided in consensus standards and in EPRI XXX *(industry document under review)*.~~

For operations in adverse environmental service conditions (EQ considerations), an evaluation[12] should be performed to provide an industrial (balance-of-plant) confidence level that the equipment will operate during such adverse environmental service conditions. Equipment operability can be established through such an engineering evaluation combined with procurement process specification and controls. Procurement requirements based on design requirements, recognized industrial/military standards, and evaluations provide sufficient confidence that the design bases functions will be satisfied during such service conditions. Standard industrial controls and procedures, e.g., licensee evaluations and vendor specifications, are sufficient, and, generally, qualification testing would not be required. Vendor activities and procedures should be reviewed, as necessary, through a licensee's commercial vendor audit program. Additional guidance is provided in Appendix D.

~~For operations in adverse environmental service conditions (EQ considerations), the use or reference to nationally recognized standards should be considered, or an evaluation should be performed to provide reasonable assurance that the function(s) required by the regulation, or by the safety analyses that are required by regulation would be satisfied in the designed service conditions. Standard commercial controls, practices and qualification procedures, are sufficient, i.e., vendor specifications or licensee analyses should be sufficient, and generally testing would not be required. Vendor activities and procedures would be reviewed, as necessary, through a licensee's commercial vendor audit program. Specifically, one or a combination of the following methods would be used. (See EPRI XXX for more detailed guidance – *document being reviewed by industry*)~~

- ~~Reference to vendor documentation (catalogues, product sheets, certificate of conformance) that indicate that the product would operate in accordance with the procurement specifications in the service conditions defined by the design~~

[11] An evaluation is defined as an analysis (traditional or computer calculations), a review of test data, a qualitative engineering evaluation, or a review of operational experience, or any combination of these elements.
[12] An evaluation is defined as an analysis (traditional or computer calculations), a review of test data, a qualitative engineering evaluation, or a review of operational experience, or any combination of these elements.

~~record file~~

~~A commercial level (BOP) equivalency evaluation that determines the procured item is equivalent to the item being replaced and will satisfy the design requirements.~~

~~A commercial level (BOP) engineering evaluation, where existing information does not provide sufficient assurance that the product would satisfy the required function(s) in the service conditions defined by the design record file.~~

QA Considerations

~~Full~~ Ccompliance with Appendix B to Part 50 is not ~~necessary or~~ required because RISC-3 SSCs are of low or no safety-significance. A licensee's industrial ~~commercial~~ (BOP) control programs are sufficient. In general, industrial ~~commercial~~ programs have similar elements to Appendix B, with less emphasis on documentation and process. Appendix C~~A~~ provides an example of the important elements that should be ~~are~~ included in a licensee's industrial ~~commercial~~ (BOP) control program.

~~As appropriate and necessary, controls, monitoring criteria, procedures and work practices are adjusted, as determined by the licensee, to take into account operating experiences and plant deficiencies. Documentation is at a level commensurate with commercial equipment and activities.~~

Application of ASME Requirements

~~Under Option 2, RISC-3 the functional design bases are not changed. However,~~ A~~a~~ licensee that chooses to include §50.55a in the matrix of regulations adopted under §50.69 would not be required by NRC regulations to apply ASME requirements to RISC-3 SSCs[13].

For those SSCs, where the licensee has analyzed the active functional significance of the SSC (e.g., Option 2 SSC categorization scheme~~surrogate modeling~~), but not the effects of the passive pressure boundary failure (i.e., indirect effects), the licensee should use the original construction code requirements or, as an alternative, other nationally recognized non nuclear Codes, Standards or Specifications suitable for that item (e.g., B31 series for piping, B16.34 for valves) in performing a repair or replacement activity on an item in that system. Use of the nationally recognized non-nuclear Codes, Standards and Specifications provides equivalence in construction and installation requirements albeit with some decreased assurance (e.g., lesser NDE, administrative requirements). *(Additional guidance is provided in the ASME Section XI, Repair/Replacement~~,~~ C~~c~~ode C~~c~~ases, which are ~~is~~ under development.)*

Alternatively, ~~In contrast,~~ a~~the~~ licensee may analyze both the ~~functional significance (i.e.,~~ direct effects (active functional failures)~~)~~ and indirect effects (passive functional failure).~~of the pressure boundary failure of the SSCs in that system.~~ If, following this additional categorization, items are ~~they conclude that the items in that system can be~~ classified as RISC-3 ~~based on this expanded~~

[13] It should be noted that while §50.55a and ASME requirements are not applicable to RISC-3 SSCs from a NRC regulatory perspective, there may be local and state requirements that may require a licensee to adopt a specific code or standard.

~~analysis,~~, and then the licensee can perform repair or replacement activities to ~~engineering specifications or~~ industry standards or licensee procedures and specifications based on industry standards. ~~developed by the licensee.~~

In instances where a licensee evaluates both active and passive effects, components, sub-components and piece-parts may be placed in different categories, e.g., the pressure boundary of a valve may be RISC-1, but the active components may be RISC-3. In these cases, ASME would govern the pressure boundary element, while other recognized industry codes or standards may govern the active function. In such cases it is important that the design record file and associated equipment databases are correctly annotated.

~~It should be noted that while §50.55a and ASME requirements are not applicable to RISC-3 SSCs from a NRC regulatory perspective, there may be local and state requirements that may require a licensee to adopt a specific code or standard~~

Application of Other Controls

Based on the reviews described in this Section, controls, monitoring criteria, procedures and work practices are adjusted, as determined by the licensee, to take into account operating experiences and plant deficiencies. Documentation is at a level commensurate with balance-of-plant equipment and activities.

10 CFR Part 21 does not apply to RISC-3 SSCs. A failure of a RISC-3 SSC, which is not safety-significant, could not result in a substantial safety hazard, a governing criteria in defining the scope of SSCs subject to Part 21.

~~10 CFR 50.59 would continue to apply to RISC-3 SSCs until the specific design bases that are linked to the RISC-3 SSCs are changed under the project for risk-informing NRC technical requirements.~~

~~For safety related SSCs that are categorized as RISC-3, yet are not directly referenced in a regulation or directly and specifically referenced in the safety analyses required by regulation, the licensee has the option of reclassifying these as RISC-4 SSCs on completion of a §50.59 evaluation.~~

~~5.6.23.3.1~~ Example 1: Low Pressure Core Spray System (RISC-3)

Prior to the adoption of §50.69, the Low-Pressure Core Spray system was categorized as safety-related and was included in the scope of the maintenance rule. In adopting §50.69, the risk-informed evaluation process classified the system as RISC-3, based on consideration of both direct and indirect effects. The analysis of direct effects led to a low safety significance conclusion because of redundancy with LPCI under realistic success criteria. A walk-through was performed and it was determined that there would be no adverse impact due to indirect effects of a failure of the pressure boundary.

The licensee's maintenance rule monitoring program established performance criteria based on all functional failure modes, not just on those associated with maintenance preventable functional failures. As a result, the licensee adopted the same reliability criteria as functional performance thresholds. A licensing commitment (part of the general commitment for RISC-3 SSCs) was made to monitor the LPCS trains to provide sufficient confidence that the design bases functions would be satisfied.~~the same functional criteria as established by the maintenance rule~~. Other commitments associated with the LPCS system were reviewed. Those commitments that were solely associated with the LPCS RISC-3 SSCs were deleted. ~~This single commitment superceded all previous commitments associated with this system.~~

The program controls were adjusted to make them consistent with the licensee ~~commercial~~industrial (BOP) activities.

The licensee selected §50.55a as one of the regulations adopted as part of the §50.69 implementation. As a result, the licensee developed a specific testing, inspection, repair and replacement program for the system which superceded the ASME Section XI and ASME O&M requirements. No other changes were made to the engineering or procurement specifications.

Subsequent to the adoption of §50.69, the system required replacement components. Replacement parts were procured to the same design engineering specifications using ~~commercial~~industrial controls and procedures. Procurement documentation included a manufacturer's certification relating to the ability of the pump to satisfy the functional performance requirements. The repairs and post-maintenance testing were carried out in accordance with ~~commercial~~industrial balance-of-plant procedures.

NOTE: If the licensee had only analyzed the direct effects of failures of SSCs in the system and a valve needed to be replaced, the replacement valve would be designed and installed to satisfy the original construction code or ANSI B16.34.

~~5.6.33.3.2~~ Example 2: Electrical Power Supply System for Containment Spray System (RISC-3)

Prior to the adoption of §50.69, the electrical system for the Containment Spray system was categorized as safety-related and was included in the scope of the maintenance rule. In adopting §50.69, the risk-informed evaluation process categorized the system as RISC-3.

In developing the performance criteria for the maintenance rule, the licensee included electrical distribution systems as a supporting element for each train. The licensee adopted the same maintenance rule performance criteria for its RISC-3 monitoring program.

With the exception of the pump motor and power cabling, the electrical system is located outside of containment in a mild environment.

For the pump-motors and cabling, work controls and procedures were changed to ~~commercial~~industrial practices. Qualification and documentation to 10 CFR 50.49 requirements and standards ~~are~~ are no longer ~~not~~ required, but ~~documented~~ vendor specifications are required and~~, where necessary,~~ analyses are~~are~~ performed to provide reasonable assurance that the equipment would~~ill~~ satisfy its design bases function ~~operate~~ in the anticipated operational environment.

In regard to the breakers and motor control switchgear, work continues to be performed using the same controls and procedures as prior to the adoption of §50.69, i.e., safety-related procedures and controls.

For spare parts, manufacturer specifications supported, where necessary, with analyses that provide reasonable assurance that the spare parts satisfy the engineering and procurement specification are ~~is~~ sufficient. Part 21 is not applicable to the cabling and motor because they are of low safety-significance and a failure could not present a substantial safety hazard.

~~The §50.59 change control process still applies.~~

~~5.6.43.3.3~~ Example 3: Hydrogen Recombiners (RISC-3)

Prior to the adoption of §50.69, the hydrogen recombiners for a PWR with a large, dry containment were categorized as safety-related and were included in the scope of the maintenance rule because they are safety-related SSCs. (The PRA and maintenance rule expert panel deliberations classified these SSCs as low risk-significant). In adopting §50.69, the risk-informed evaluation process classified the hydrogen recombiners as RISC-3 because their loss would not impact the plant safety ~~risk~~ profile in terms of CDF or LERF. Additionally, loss of this function would not have impacted the plant safety functions, nor would it have contributed to a credible core damage or a release of fission products.

The licensee's maintenance rule monitoring program established performance criteria based on all failure modes, not just on those associated with maintenance preventable functional failures. ~~As such, T~~The licensee adopted the ~~same~~ functional criteria developed to support the maintenance rule reliability determinations for §50.69.

A licensing commitment (part of the general commitment for RISC-3 SSCs) was made to monitor the hydrogen recombiners to provide sufficient confidence, as demonstrated by the satisfaction of the functional criteria, that the design bases functions would be satisfied.~~the same functional criteria as that established by §50.65. This single commitment superceded all previous commitments.~~

Program controls were adjusted to make them consistent with standard balance-of-plant activities. Electrical controls and work practices were adjusted to those of the licensee's ~~commercial~~industrial (BOP) programs up to the first isolation device. For spare parts, manufacturer certification that the spare parts satisfy the engineering and procurement specification is sufficient. Part 21 is not applicable because the SSCs are of low safety-significance and a failure of the SSC could not present a substantial safety hazard.

~~Following the issuance of a final rule on §50.44, which deleted the requirement for hydrogen recombiners in large dry containments, the licensee performed a §50.59 to reclassify these SSCs as RISC-4.~~

~~5.7~~6.4 ~~3.4~~ TREATMENT OF RISK-INFORMED SAFETY CLASS 4 SSCs

Risk-Informed Safety Class 4 SSCs are categorized as not being safety-significant and are not safety-related. These SSCs are not subject to NRC regulations[14].

RISC-4 SSCs may include some nonsafety-related, important-to-safety SSCs that are governed by regulations, such as, 10 CFR 50.49, *Environmental qualification of electric equipment important to safety for nuclear power plants;* 10 CFR 50.62, *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants;* and 10 CFR 50.63, *Loss of all alternating current power.* Design bases functionality is still necessary although industrial level controls may be substituted for the "augment quality controls", defined in regulatory guidance documents, such as Regulatory Guide 1.155. As such, these SSCs would be subject to functional monitoring to assure that the design bases functions will be satisfied.

Depending upon circumstances, the RISC-4 monitoring criteria may be eliminated through the application of a §50.59 evaluation.

~~NOTE: This category of SSCs is included in the scope of NRC oversight programs to the extent that a failure of a RISC-4 SSC degrades a safety-significant (RISC-1 or RISC-2) structure, system or component to the extent that the associated safety-significant function cannot be satisfied.~~

[14] This category of SSCs is included in the scope of NRC oversight programs if a failure of a RISC 4 structure, system or component resulted in a failure of a safety-significant functional requirement.

7 CONTROL PROCESSES FOR LICENSEES ADOPTING 10 CFR 50.69

7.1 APPLICATION OF 10 CFR 50.59

10 CFR 50.59 continues to be applied to facility changes. In many cases the change could be screened out because the change does not degrade the design bases.

7.2 CHANGE CONTROL PROCESS FOR SAFETY-SIGNIFICANT BEYOND DESIGN BASES FUNCTIONS

The §50.59 process screening criteria focuses its change control activities on matters that could affect a design function[15]. The §50.59 change control process does not fully evaluate changes that effect safety-significant beyond design bases functions. As a result, a licensee that chooses to adopt §50.69 should amend its configuration control process to include a provision that provides reasonable assurance that the safety-significant beyond design bases function(s) will be satisfied following a facility change. This additional control provision is not part of the §50.59 process.

The design control (change) element in the configuration control program is not changed and continues to ensure that the design is controlled and maintained. The additional change control provision determination should be based on evaluations (quantitative or qualitative), or on a combined quantitative and qualitative evaluation of the change and how it impacts the original beyond design bases function(s). The information contained in the modification package, the risk-informed categorization process, and the design record file, provide the detailed basis for the evaluation. Each proposed change package should be supported by engineering information, that may include but is not limited to, drawings, specifications, narrative description, design evaluations, installation and testing requirements, associated procedure changes (if any), revised analyses (if any) and similar information. This information demonstrates the safety and effectiveness of the change and is the mechanism for management approval of the implementation.

If the change control evaluation or licensee management reviews conclude that there is insufficient assurance that the "beyond design basis" safety function would be satisfied following the implementation of a change, a licensee takes the following action:

[15] As used in this section, the term function or design function relates to the interpretation provided in NEI 96-07, *Guidelines for 10 CFR 50.59 Safety Evaluations*, i.e., design bases functions or design functions that directly support the design bases functions as defined in NEI 97-04, *Design Bases Program Guidelines.*

(i) Assess the impact on the SSC categorization and the plant's risk management profile (PRA)[16] of not making the change, or
(ii) Amends the proposed change so that the above criteria are satisfied.

If the change results in a change of RISC categorization, the NRC is notified

Design record files and the PRA are updated and the NRC would be notified of changes in SSC categorization. Any changes to the UFSAR would be made in accordance with §50.71(e) and NEI 98-03, Rev. 1, *Guidelines for Updating Final Safety Analysis Reports.*

The engineering and operations documents associated with RISC-1 SSCs are already included in the scope of controlled documents for the plant. Information and action taken in response to the implementation of §50.69 relating to "beyond design bases" conditions should be documented in the engineering record files.

7.3 CHANGES TO COMMITMENTS

Changes to NRC commitments associated with any RISC SSC category should be controlled through NEI 99-04, Rev 1, *Guidelines for Managing NRC Commitment Changes*, which has been revised to reflect the impact of §50.69.

7.4 CHANGES TO SSC CATEGORIZATION PROCESS

The risk-informed §50.69 SSC categorization process should be documented in a licensee controlled document. In a licensee's §50.69 NRC submittal, a licensee makes a commitment to inform the NRC of changes in the categorization of SSCs, and to update the PRA at periodic intervals based on the ASME PRA Standard (See Section 7.0).

In accordance with NEI 98-03, *Guidelines for Updating UFSARs*, the categorization process should be described in a licensee's controlled document, not in the UFSAR. Changes to the categorization process should be controlled through the application of the NRC commitment management process, as described in the NRC endorsed NEI 99-04, *Guidelines for Managing NRC Commitment Changes*. The UFSAR guideline has been has been revised to reflect a risk-informed regulatory regime, such as §50.69. Changes in the PRA that result in changes in SSC categorization should be reported to the NRC at intervals consistent with the UFSAR updates.

Changes to the Plant Specific PRA

The plant specific PRA should be maintained and upgraded, such that its representation of the as-built, as-operated plant is sufficient to support applications for which it being used.

[16] The effect of the change or not making the change could result in a change in SSC categorization of the SSCs directly related to the proposed change as well as other SSCs that are not related to the proposed change.

A licensee's configuration control program should monitor changes in the design, operations, maintenance and industrywide operating experience that could affect the plant and the PRA. The program should include monitoring of changes in PRA technology and industry experience that could change the results of the PRA model.

Changes to the plant specific PRA should be reviewed for potential changes to the SSC categorization.

~~7.4~~7.5 CHANGES TO SSC TREATMENT

Changes to NRC Special Treatment Requirements for RISC-1 SSCs

Changes to the existing NRC special treatment requirements for RISC-1 SSCs continue to be subject to §50.59 and §50.54(a).

Changes to Licensee Industrial (BOP) Controls for RISC-2, RISC-3 and RISC-4 SSCs

Changes to a licensee's industrial (balance-of-plant) of plant controls and augment quality controls would be governed by a process similar to that for changes to a licensee's quality assurance program description.

A licensee's industrial level control program description should be described in the UFSAR. Appendix C provides an example of such a program description.

Changes to the industrial level (BOP) program is controlled through the licensee's configuration control program and through the same mechanism that control's a licensee Part 50, Appendix B Quality Assurance program. Prior NRC review and approval would be required when the change results in a reduction in commitment. The commitment in this case would be associated with the program description described in the UFSAR. If the proposed change would not result in a reduced program, as defined by the existing program description in the UFSAR the change would be implemented without prior NRC review and approval.

Changes and updates to the balance-of-plant industrial controls program should be part of the periodic UFSAR update in accordance with 10 CFR 50.71(e).

~~7~~8 DOCUMENTATION & APPROVAL

To facilitate the NRC staff's review to ensure that the analyses conducted were sufficient to conclude that the key principles of risk-informed regulation have been met, documentation of the evaluation process and findings are expected to be maintained. The integrated decision process should be documented to include, descriptions and justifications of deviations from this guidance, references to sources of information and data, assumptions, limitations, weighting factors relative to operating modes and risk sources, decision tools applied, analytical techniques, resolution of conflicts between deterministic and risk evaluation results, resolution of differences of expert judgement, complete description of evaluation results, and performance monitoring program. Documentation will also include procedures that govern the integrated decision process including specifications on the IDP and its activities.

The following shall be documented and available for NRC review:

- Results of the relative risk importance of SSCs modeled in the PRA including the results of sensitivity analyses.

- Results of the final SSC categorization including a summary of IDP deliberations for each safety-related SSC classified as low safety significant and each non-safety-related SSC classified as safety significant. Decision criteria in terms of qualitative assessments, assessments for initiating events and plant operating modes not modeled in the PRA, defense-in-depth, and safety margins must be included. Technical basis documents used to support the categorization shall also be available. For safety-related SSCs which are classified as RISC-1, i.e., their classification is unchanged and no new safety significant attributes have been identified, existing documentation is sufficient and does not need to be revised.

- Functional requirements for each SSC receiving revised treatment, the original treatment requirements for these SSCs, the revised requirements for these SSCs, target values for SSC reliability and availability, and the process that will be used to assure these functional requirements and target values will be preserved/met.

- The assessment (qualitative and/or quantitative) of the overall change in plant risk as a result of changes in treatment requirements, including the baseline CDF and LERF and the change in this CDF and LERF.

  Requirements for the IDP including, the plant procedure, expertise, membership, training, and decision-making guidelines. Meeting minutes should also be included.

The basis for the IDP decisions on categorization would be part of the controlled engineering record files for the system. The file would be updated in accordance with licensee configuration control practices and would be one of the documents reviewed in the development of a design change package or when the PRA is updated per the guidance in industry standards and licensee procedures.

- The PRA and other supporting analyses, together with a description of justification of the quality and applicability of these analyses.

This documentation should be maintained by the licensee, as a controlled record, so that it is available for examination. Documentation of the analyses conducted to support changes should be maintained as lifetime quality records in accordance with Regulatory Guide 1.33.

NRC Review and Approval

As per 10 CFR 50.69, a licensee wishing to adopt a risk-informed SSC scope for special treatment requirements will make a submittal to the ~~notify the~~ Commission ~~in writing of its intent to implement this voluntary option. The notification letter will list:~~ for NRC review and approval for adopting §50.69. Appendix B provides an outline of a submittal.

~~the regulations being adopted;~~
~~the implementing methodology; and~~
~~a general schedule for implementation.~~

~~If the risk-informed evaluation methodology is different from that described in this guideline, the notification letter will include a copy of the licensee's risk-informed methodology. The notification and, where applicable, the methodology will be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office (NRR) or 60 days after submittal to the Commission, whichever occurs first.~~

Periodic Review

Changes in PRA inputs or discovery of new information described in the above paragraphs should be evaluated to determine whether such information warrants PRA maintenance or upgrade.

Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are incorporated as soon as practical.

Following revisions or updates to the PRA a review of the SSC categorization should be performed. Such reviews should include:

- A review of the PRA
- A review of plant modifications since the last review
- A review of plant specific operating experience that could impact the SSC categorization.
- A senior management review of the results
- A review of the importance measures used for screening in the categorization process[17]

Additional guidance on the update of PRAs is provided in Section 5 of the draft ASME PRA Standard.

~~At intervals not exceeding 36 months, or when appropriate, a licensee should conduct a review of the SSC categorization to take into account operating experience (industry-wide and plant-specific), risk insights, and plant modifications. The review should determine the necessity of updating the list of safety-significant SSCs. The review should encompass the following elements:~~

~~Review of, or update of the plant specific PRA to reflect changes in plant configuration, operations, and plant specific operating experience. If generic industry data has been used in the risk-informed evaluation process, then a review of industry operating experience and other pertinent databases should be performed.~~

~~Review of changes to plant activities that could impact the categorization results.~~

~~Review of plant specific operating experience and data that could impact the categorization results.~~

~~Assessment of the impact of the three elements listed above on the risk-informed SSC categorization by the Integrated Decision Process Panel~~

~~Recommendations to change categorization.~~

UFSAR

A Licensee that adopts §50.69 should update its UFSAR on completion of implementing treatment to the first set of systems that have been selected. The update should be performed in accordance with NEI 98-03, *Guidelines for Updating Final Safety Analysis Reports.* The update would include the program description of the industrial level (balance-of-plant) treatment controls. Appendix C provides such a description.

[17] If a review of the importance measures indicate that the SSC should be reclassified then both the relative and absolute values of the risk metrics should be considered by the review entity and the senior management review group.

89 ~~5~~ REFERENCES

1. EPRI TR-105396, *PSA Applications Guide*,
2. Regulatory Guide 1.174, *An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*,
3. NRC SECY 99-256, *Rulemaking Plan For Risk-Informing Special Treatment Requirements*,
4. NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*
5. NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*
6. NRC Regulatory Guides 1.175, 1.176, 1.177 and 1.178,
7. ASME Code Case OMN-3, *Requirements for Safety Significance Categorization of Components using Risk-Insights for Inservice Testing of LWR Power Plants*
8. Nuclear Energy Institute, "NEI 00-02, Revision 3, *Probabilistic Risk Assessment (PRA) Peer Review Process Guidance*," ~~Revision A3~~.
9. NEI 96-07, *Guidelines for 10 CFR 50.59 Safety Evaluations*
10. NEI 97-04, Revision 1, *Design Bases Program Guidelines*
11. NEI 98-03, *Guidelines for Updating Final Safety Analysis Reports*
12. ASME Code Case N-XXX *Alternative Repair/Replacement Requirements for Structures, Systems and Components Classified in accordance with Risk-Informed Processes*
13. ASME Code Case, X-XXX, *Risk-Informed Safety Classification of SSCs Pressure Boundaries*

APPENDIX A

GLOSSARY

Beyond design bases functions are those functional requirements that have been identified by a risk-informed evaluation process as being safety-significant yet are not encompassed by the original licensing basis for the facility

Common cause failure (CCF) – a failure of two or more components during a short period of time as a result of a single shared cause *(ASME PRA Standard)*

Core damage – uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release *(ASME PRA Standard)*

Core damage frequency (CDF) – expected number of core damage events per unit of time. *(ASME PRA Standard)*

Defense-in-depth is the application of deterministic design and operational features that compensate for events that have a high degree of uncertainty with significant consequences to public health and safety.

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. *(10 CFR 50.2)*

Design functions are UFSAR-described design bases functions and other SSC functions described in the UFSAR that support or impact design bases functions. *(NEI 96-07)*

Design bases functions are functions performed by systems, structures and components (SSCs) that are (1) required by, or otherwise necessary to comply with, regulations, license conditions, orders, or technical specifications, or (2) credited in licensee safety analyses to meet NRC requirements. *(NEI 97-04)*

Dependency – requirement external to an item and upon which its function depends *(ASME PRA Standard)*

Diverse – replication of an activity or structural, system, train or component requirement using a different design or method.

Evaluation is defined as an analysis (traditional or computer calculations), a review of test data, a qualitative engineering evaluation, or a review of operational experience, or any combination of these elements. *(Industry UFSAR s)*

Fussell-Vesely (FV) importance measure – for a specified basic event, Fussell-Vesely importance is the fractional contribution to the total of a selected figure of merit for all accident sequences containing that basic event. For PRA quantification methods that include non-minimal cutsets and success probabilities, the Fussell-Vesely is calculated by determining the fractional reduction in the total figure of merit brought about by setting the probability of the basic event to zero. *(ASME PRA Standard)*

Large early release – the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions *(ASME PRA Standard)*

Large early release frequency (LERF) – expected number of large early releases per unit of time *(ASME PRA Standard)*

Probabilistic risk assessment (PRA) – a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment, PSA) *(ASME PRA Standard)*

Redundant – duplication of a structure, system, train, or component to provide an alternative functional ability in the event of a failure of the original structure, system, train or component

Risk -- Risk encompasses what can happen (scenario), its likelihood (probability), and its level of damage (consequences). *(NUMARC 93-01, Rev 2)*

Risk achievement worth (RAW) importance measure – for a specified basic event, risk achievement worth importance reflects the increase in a selected figure of merit when an SSC is assumed to be unable to perform its function due to testing, maintenance, or failure. It is the ratio or interval of the figure of merit, evaluated with the SSC's basic event probability set to one, to the base case figure of merit. *(ASME PRA Standard)*

Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures set forth in §50.34(a)(1) or §100.11 of this chapter, as applicable. *(10 CFR 50.2)*

Safety-Significant structures, systems and components are those structures, systems and components that are significant contributors to safety as identified through a blended risk-informed process that combines PRA insights, operating experience and new technical information using expert panel evaluations.

Severe accident – an accident that usually involves extensive core damage and fission product release into the reactor vessel, containment, or the environment.

Train A collection of equipment that is configured and operated to serve some specific plant safety function and may be a sub-set of a system. The utility can utilize the FSAR or PRA analysis to better define the intended configuration and function(s). *(NUMARC 93-01, Rev 2)*

~~commercialcommercialcommercial. They are provided for information, not guidance.~~

~~incorporates the procedure for implementing changes associated with safety-significant "beyond design basis" changescommercial~~

APPENDIX B

SUBMITTAL OUTLINE

Table of Contents

1. Introduction

This section would provide a statement of the objective of the submittal and identify the unit(s) included in the Option 2 submittal. It may also include a general statement of the approach to be taken, the general scope and the anticipated schedule.

2. SSC Scope & Approach

This section would provide an overview of the approach taken including any exceptions or supplements to the NEI & regulatory guidance. In addition, this section should include a definition of the scope of the special treatment requirements being modified.

2.1 Safety-Related SSCs

This section would describe the scope of safety-related SSCs to be considered in the categorization process.

2.2 Non-~~Safety Related~~Safety-related SSCs

This section would describe the scope of non-safety-related SSCs to be considered in the categorization process.

2.3 Schedule for Implementation

This section would provide the anticipated schedule for the categorization effort and the schedule for the implementation of changes to the special treatment requirements.

3. Categorization Basis

This section would provide a summary of the categorization bases to be used.

———3.1 Plant-Specific Risk Information

This section would describe the specific risk analyses to be utilized, the basis for determining that those analyses are both applicable and useful in categorization.

———3.2 Characterization of PRA Quality

This section provides the basis for determining that the risk information utilized in the categorization is technically capable of supporting the categorization process. The following information would be included:

Internal Events PRA
- A basis for why the internal events PRA reflects the as-built, as-operated plant.
- A high level summary of the results of the PRA peer review~~certification~~ of the internal events PRA including elements which received grades lower than 3.
- The disposition of any peer review fact and observations (F&Os) classified as A or B importance.
- Identification of and basis for any sensitivity analyses necessary to address identified elements and F&Os.

Other PRA Analyses
- A basis for why the other PRA analyses adequately reflect the as-built, as-operated plant.
- A disposition of the impact of the significant peer review findings on the other PRA analyses.
- Identification of and basis for any sensitivity analyses necessary to address issues identified in the other PRAs.

4. Integrated Decision--M~~m~~aking Panel

This section would provide a summary of the IDP process to be used.

4.1 Panel Makeup

This section would describe the makeup of the IDP:
- Plant Operations (SRO qualified),
- Design Engineering (including safety analyses),
- Systems Engineering,
- Licensing, and
- Probabilistic Risk Assessment.

In addition, the approach to training the IDP would be described.

4.2 IDP Guidance

This section would provide a summary of the plant-specific IDP procedures to be used including the approach to documenting the IDP's recommendations on changing the initial categorization of an SSC.

5. Treatment

This section would provide a summary of the changes in special treatment requirements expected from the categorization.

5.1 ~~Commercial~~Industrial Program Summary

This section would provide a summary of the ~~commercial~~industrial program to be applied to RISC-2 and RISC-3 SSCs.

6. Documentation Update

This section would describe the licensee's approach to updating the documentation necessitated by the categorization, including any UFSAR changes anticipated.

7. Change Control Process

This section would provide a summary of change control process to be used after the changes in special treatment have been made. In addition, this section will describe the licensee's approach to periodic reviews and updates of the categorization and treatment.

8. References

This section would provide a list of the key references expected to be used.

APPENDIX C

Examples of Program Elements in a Licensee's Industrial Control Program for RISC-2, RISC-3, and RISC-4 SSCs

Introduction

Many plants do not have a specific procedure of program labeled "industrial quality program." Rather, such programs and procedures are disseminated in numerous plant programs and procedures. When combined together, as a whole, these program elements assure that the proposed industrial treatment provides reasonable assurance that the RISC-2 safety-significant functions and the RISC-3 required (safety and regulatory) functions will be satisfied. . These programs are currently in place, and provide an effective means of addressing the special treatment controls for RISC-2 and RISC-3 SSCs. In many instances, such programs and procedures are a subset of the more formal 10 CFR 50, Appendix B quality programs

The following control element summaries are the central and important segments of a typical licensee's industrial control program

I. Monitoring and Assessment Program

Monitors structures systems and components to provide reasonable assurance that the safety-significant, power production and required regulatory functions will be satisfied. It provides input into the facility assessment programs such as the ~~and~~ maintenance rule, 10 CFR 50.65, and erosion and corrosion programs.

Assessments are implemented to provide adequate assurance that the performance criteria and processes are being achieved and implemented effectively. The type, frequency and degree of specificity of assessments are determined by the importance to the safety functions and the performance history of structures, systems, components, or the work activity being evaluated.

Assessments may be in the form of reviews, monitoring, tests, surveillances, inspections, audits or examinations, as appropriate. These assessments are performed by line organizations or personnel, by management, or by independent internal or external organizations or groups. The importance to the safety function and performance history determines the degree of management and technical oversight. Personnel performing assessments are qualified through training, work experience, or certification.

II. Corrective Action Program

Defects and deviations from the prescribed performance criteria or work processes are identified and communicated to the appropriate levels of management for corrective action, in a timely manner. When necessary, controls and processes are available to stop work while the appropriate level of management resolves a deviation or concern. Satisfactory accomplishment of corrective actions shall be confirmed by the appropriate level of line management consistent with the importance of the defect or deficiency.

Evaluation of Deviations

Documented deviations from design specifications, performance criteria or work processes are evaluated commensurate with the importance to the safety significant functions, power production goals, and personnel safety. As appropriate and commensurate with the importance of the defect or deficiency, the evaluation considers the cause of the deviation, the significance and extent of the defect or deficiency in the work activity, with input from the appropriate personnel associated with the activity in which the deviation was identified

Resolution of Deviations

Documented deviations shall be resolved by the responsible organizations to an extent, and in a manner, that is consistent with the importance of the structure, system, component or activity. Activities associated with correcting deviations shall continue until the performance criteria have been satisfied, or until appropriate levels of management justify and authorize changes to the original performance criteria.

III. Maintenance Program

Incorporates the requirements to support 10 CFR 50.65 and includes the preventative maintenance (PM) and the predictive maintenance program.

A. Maintenance Rule Program

Implements the Maintenance Rule at the station, including SSC scoping and monitoring, classifying SSC performance in accordance with criteria and goals, ensuring proper corrective actions when performance criteria are not met, and periodically evaluating overall program performance.

Note: The maintenance rule program provides a basis for the performance monitoring program for RISC-3 SSCs

B. The Preventive Maintenance (PM)

Program provides for the identification, scheduling, and assessment of routine preventive maintenance activities on RISC-4, RISC-3 and, where appropriate, RISC-2 SSCs. The PM program focuses on maintenance activities that assure SSCs will continue to satisfy the designed functions. As appropriate, PM activities encompass important design elements, historical performance, and established maintenance practices. PM activities include, where appropriate, routine maintenance checks, inspections, replacements, tests, adjustments, and calibrations. The program is adjusted, as necessary, based on the results of the PM program. If a deficiency cannot be corrected under a PM activity, then action is taken in accordance with the Corrective Action Program until the deficiency is resolved. When necessary, post-maintenance testing is performed prior to returning equipment to service.

C. Predictive Maintenance Program

The Predictive Maintenance Program provides for periodic, proactive testing of selected SSCs to identify a decline in performance or in material condition. Predictive maintenance activities assist in assuring that SSCs continue to perform reliably and provide additional confidence that the SSC design functional requirements will be available when required. Activities include: periodic lube oil analyses on large motors and pumps; vibration analyses of rotating equipment; thermographic analyses of both mechanical and electrical SSCs to identify improper temperature conditions or electrical hot spots; acoustic analysis for valve leak-by or SSC leakage; and motor potential diagnostic testing. Deficiencies identified through the Predictive Maintenance Program are resolved through the Corrective Action Program.

IV Configuration Control Program

Manages and controls changes (procedural and equipment) to the facility to assure the plant configuration and practices correctly reflect the design record file and licensing documentation. The program includes the §50.59 change control process and the program for managing NRC commitments.

Work Planning and Schedule

This program provides the requirements and guidelines for planning and scheduling maintenance and other work activities to optimize plant operational safety, reliability and availability. The program addresses the planning and scheduling of the following activities:

Corrective, preventive, and pre-determined (i.e., planned or repetitive) maintenance

On-line maintenance

Periodic testing

Installation of design change packages

Design Change Program

Establishes the process for managing the preparation, implementation, and where necessary, the licensing of design changes to SSCs. It defines the controls necessary to ensure safe implementation of station design changes and provides reasonable, industrial level assurance that changes to the facility are implemented consistent with the information contained in the plant's design record file. As necessary and appropriate, post-modification testing is performed to determine or verify the capability of a modified SSC to meet specified functional design requirements and design bases before being placed in service.

The design change process for RISC-2 SSCs includes a provision for assessing and managing the change in risk from equipment design changes.

If spare parts are not available from the original equipment manufacturer, an engineering evaluation is performed to determine the applicability of alternative suppliers. The evaluation assesses the functional differences associated with fit, form, function, and conditions of service of the equipment or service being supplied.

Procedure Program

This element applies to technical and administrative procedures and includes the necessary processes to maintain procedure quality. The program further establishes the processes for 1) the development, review, and approval of new procedures, procedure revisions, procedure changes and procedure deletions, 2) review and approval of vendor procedures. The program is designed to assure consistency in the development of new procedures, and in the review and approval of procedure changes.

~~5.4.3~~V Procurement Program

Procurement of SSCs is controlled by administrative procedures that implement quality assurance program elements for procurement and materials management consistent with safety and power generation. These procedures provide reasonable assurance that the procurement specifications reflect the appropriate requirements of the design record file. As necessary, and consistent with the safety-significance or power production requirement, the program includes: vendor surveillance audits and maintenance of approved vendor lists, receipt inspection, materials verification activities, special handling and storage procedures that are consistent with the information in the design record file.

The procurement specification includes engineering specifications that reflect the design record file requirements and include service condition parameters.

APPENDIX D

EVALUATION AND PROCUREMENT ~~SPECIFICATION~~ OF ITEMS IN RISC-3 AND RISC-4 APPLICATIONS PREVIOUSLY GOVERNED BY 10 CFR 50.49

The application of NRC 10 CFR 50.49, ~~Environmental Qualification (EQ)~~ special treatment requirements ~~in the procurement process~~ provides increased assurance, above that normally provided by industrial (licensee balance-of-plant) processes, that the safety functions will be satisfied under designed service conditions. Prior to ~~the~~THE §50.69 categorization of SSCs, §50.49 requirements were applied to applicable safety-related SSCs and specific nonsafety-related SSCs whose failure could impact a safety-related function.

~~Under §50.69, while special treatment requirements are applied consistent with safety significance to provide increased assurance of functionality, the design bases are not changed. As a result, EQ requirements continue to be applied to high safety-significant, safety-related SSCs (RISC-1 SSCs) and specific nonsafety-related, safety-significant SSCs (RISC-2 SSCs) that were previously the subject of §50.49 controls.~~

Under §50.69, while special treatment requirements are applied consistent with safety significance to provide assurance of functionality, the design bases are not changed. As a result, the qualification methodologies specified by §50.49 SHOULD continue to be applied to high safety-significant, safety-related SSCs (RISC-1 SSCs) and specific nonsafety-related, safety-significant SSCs (RISC-2 SSCs) that were previously the subject of §50.49 controls. However, alternative methods (i.e., industrial level controls) may be applied to low safety-significant RISC-3 and RISC-4 items previously within the scope of §50.49 to provide sufficient confidence that the equipment will function in its designed service conditions.

RISC-3 SSCs are not safety-significant, yet they continue to be labeled as safety-related until the design bases are changed. As such, from a licensing perspective, there is a need to provide a level of assurance (an industrial level) that the designed function will be satisfied under the designed service condition. In view of the lower safety significance the level of assurance can be lower than that provided for safety-related equipment. The application of industrial level controls provides the necessary degree of assurance for this equipment. Examples of typical elements in a licensee's industrial (BOP) level controls programs are provided in Appendix C~~A~~. These BOP industrial controls, which include design, procurement, configuration control, and maintenance, when coupled with engineering knowledge and operating experience, provide an industrial level of assurance that the equipment will function in its designed service conditions.

Two ~~of these~~ elements in an industrial level control program are design and procurement. The aim of these design and procurement measures and controls is to assure that the design is capable of performing the required function and that the purchased items meet the design requirements. Qualification is the verification of design limited to demonstrating that the electrical equipment is capable of performing required functions under harsh environmental conditions[18]. Adequate confidence that the design bases functions will be satisfied during such service conditions for RISC-3 equipment can be achieved through an engineering evaluation, performed as part of the design or procurement process, combined with procurement requirements based on design requirements, the evaluation conclusions, and nationally recognized industrial/military standards. ~~Another element is the preventative maintenance program, which determines the appropriate replacement intervals equipment.~~

~~Specification and~~ Replacement of RISC-3 and RISC-4 SSCs that were formerly subject to 10 CFR 50.49

The ~~procurement process~~ procurement process consists of three distinct, yet related processes—the design process, the technical evaluation process and the acceptance process.

- The design process results in a set of design requirements and parameters[19].
- The technical evaluation process translates the design requirements ~~criteria~~ into procurement specifications and, where necessary, forms the basis for acceptance criteria. From these design requirements, existing equipment ~~is~~ specification and from experience with procuring environmentally qualified components, licensees can identify the important aspects and material requirements associated with the providing sufficient confidence that the equipment should satisfy its design bases functions in its designed service environment.~~design requirements~~.
- The acceptance process assures the item received conforms to requirements specified in the acceptance criteria or procurement document(s).

Equipment can be replaced ~~is changed out on an as-needed basis because of wear, defects, or preventative maintenance program considerations. RISC-3 equipment can be replaced~~ in three ways:

- Identical replacements,
- Equivalent replacements, or
- New equipment design.

In each of these cases the equipment need not be procured from the same vendor. ~~No matter the option, the BOP industrial controls, which include design, procurement, configuration control, and maintenance, when coupled with engineering knowledge and operating experience provide an industrial level of assurance that the equipment~~

[18] Reg. Guide 1.189, Rev. 1

[19] Design requirements and parameters include the service conditions, which the equipment will experience during operation. Service conditions include temperature, pressure, humidity, chemical effects, water immersion and radiation.

~~will function in its designed service condition. In each of these cases the part need not be procured from the same vendor.~~ In addition, the vendor's quality program would not have to satisfy Appendix B to Part 50 and 10 CFR Part 21 would not be applied.

Identical Replacement

This refers to circumstances that involve identical design or configuration~~. No engineering evaluation is necessary. The suitability of the design has been previously established.~~

Identical replacements may include items which have part/model number differences because of administrative changes, identical items purchased from alternate or sub-tier suppliers and items manufactured to industry standards but purchased from an alternate supplier. There is no need for additional qualification documentation, including component test documentation, material certification or vendor audits. A~~, but a~~ licensee may impose such additional controls on a case-by-case basis linked to the supplier's and equipment performance histories.

Design

The suitability of the design has been previously established.

Procurement

1. The procurement document should specify sufficient detail to ensure the replacement item is technically identical to the original. On a case-by-case basis and based on past procurement history with the vendor, the licensee may decide to contact the vendor to confirm that there has been no change to the design or materials, even if the part number remains the same.
2. Receipt inspection should verify the correct item was received (e.g., check part number and configuration). The receipt inspection should include a review of any documentation requirements imposed by the purchase order.

~~***Example***~~

Equivalent Replacement

The design or configuration is not the same, yet the design is not significantly different.

Design

The suitability of the design has been established except for those areas where minor differences in design or configuration have been identified. A~~The~~ licensee should perform~~s~~ a technical evaluation, similar to an equivalency evaluation, consistent with the controls and practices for balance-of-plant equipment to assess the effect that differences in the design would have on the ability of the item to perform its design~~ed~~ function in its designed~~ed~~ service environment. There is no need for additional documentation, such as component test documentation, material certification or vendor audits, but a licensee may impose such additional controls on a case-by-case basis linked to the supplier's and equipment performance histories. ~~For items determined to be equivalent, suitability of the design has been previously established.~~

Equivalency may be established using vendor documentation, including documented telephone calls, documented engineering judgement, operating experience, and other available data sources, existing qualification reports, and existing industrial material data.

Procurement

1. The procurement document should specify sufficient detail to ensure the replacement item is within the evaluated differences.
2. Receipt inspection should verify the correct item was received (e.g., check part number and configuration). The receipt inspection should include a review of any documentation requirements imposed by the purchase order.

   ~~*Example*~~

New Equipment Design

The design or configuration is not the same and the licensee has determined that a design change package should be developed. This process sh~~w~~ould be used when the design is being changed because there are substantial design differences between the original item and the replacement item, or when the items do not satisfy the equivalency determination of the previous section.

Design

~~In these cases additional~~ Eengineering evaluations should be performed ~~are necessary~~ to establish that the design is suitable for the application. The process followed sh~~w~~ould be governed by the licensee's design change and configuration control procedures. New design and procurement specifications should be~~are~~ established for the replacement item. The need for specific testing and validation should be~~is~~

determined on a case-by-case basis, as determined by a licensee's engineering and procurement groups.

The type and complexity of any evaluations and the need for testing should be~~is~~ dependent on the severity of the environment as well as the equipment type.

Procurement

1. The procurement document should specify sufficient technical detail to ensure the replacement item meets the evaluation requirements. The need for documentation or supplier assessment is based on the new design specification and supplier performance history.
2. Receipt inspection should verify the correct item was received and documentation should be reviewed against the documentation requirements of the purchase order.

~~*Example*~~

General Examples[18]

The scope and details of the engineering evaluation will typically be related to the severity of the environmental conditions, along with the equipment type, desired functions, and available performance information. For most applications, the environmental conditions will fall into one of the following categories:

- Radiation Only DBE Environment
- Thermal Only DBE Environment
- Condensing Moisture HELB (Fast Transient, Limited Thermal Energy)
- HELB with Significant Thermal Content
- LOCA/MSLB

Radiation Only DBE Environment—this type of environment has a significant change in radiation conditions only. Depending on the severity of the radiation dose the following options should be considered:

- <10 Kilorad—no engineering evaluation is necessary to address environmental parameters. The radiation level is not sufficient to adversely affect equipment or materials. Procure and accept industrial equipment as described above.
- <1 Megarad—E~~e~~xclude use of Teflon and other specific fluorocarbon materials adversely affected by the radiation level. A

[18] (All of the examples exclude components containing electronic devices, e.g. digital. The ability of electronics to withstand these environments would be evaluated on a case by case basis; however, consideration may be possible on a class or type basis.)

licensee should use published information on material radiation resistance. No other environmental issues need to be ~~must~~ be addressed. Procure and accept industrial equipment as described above.

- >I Megarad—An evaluation should be performed to evaluate radiation capability of the materials through industry databases and other industry documents, where available.~~, such as, EPRI NP 4172, Radiation Data for Design and Qualification of Nuclear Power Plant Equipment.~~ If existing data is insufficient to assure acceptable performance, ~~perform~~ radiation testing should be performed to required level. No other environmental issues need~~must~~ be addressed. Procure and accept industrial equipment as described above.

Thermal Only DBE Environment—this type of environment involves an increase in the operating temperature of the equipment that is significantly different that the operating temperature during normal operation, including anticipated operational transients. Such an increase may occur due to assumed unavailability of certain HVAC equipment or due to additional heat loads in a plant area. Depending on the severity of the temperature change the following options should be considered:

- Specify equipment with appropriate temperature rating as determined through review of vendor documentation. Procure and accept industrial equipment as described above.
- If vendor documentation does not adequately address the required ratings, evaluate thermal capabilities using information from industry databases. Procure and accept industrial equipment as described above.
- If existing data is insufficient to assure functionality, verify ~~perform~~ operability at the specified temperatures using industrial-type testing.~~thermal operability testing~~. Procure and accept industrial equipment as described above.

Condensing Moisture HELB (Fast Transient, Limited Thermal Energy)—this type of environment involves moderate temperature increases combined with condensing moisture from a short duration HELB. These conditions are typical for many outside containment areas that are somewhat removed from but communicate with HELB plant areas. Depending on the severity of the temperature and moisture changes, the following options should be considered:

- Evaluate susceptibility of equipment to condensing moisture. Evaluate moisture protection capability provided by the equipment housing. ~~housing if equipment is located inside a structure.~~ If equipment is protected from moisture and condensation drains away from the equipment, procure and accept industrial equipment as described above. Consider radiation effects as in “radiation only” DBE environment.

- If directly exposed to moisture, specify sealed components or components designed for high humidity or water spray conditions, (e.g., or jungle-rated equipment). Procure and accept industrial equipment as described above. Consider radiation effects as in "radiation only" DBE environment
- If existing data is insufficient to assure functionality, verify operability at the specified temperature and moisture conditions using industrial-type testing. Procure and accept industrial equipment as described above.

HELB with Significant Thermal Content—this type of environment typically occurs in plant areas experiencing the direct effects of HELBs. Depending on the severity of the temperature, pressure, and moisture changes, the following options should be considered.

- Evaluate susceptibility of equipment to condensing moisture. Evaluate the protection provided by the equipment housing.~~housing if equipment is located inside a structure.~~ If equipment is protected from moisture and condensation drains away from the equipment, evaluate thermal capability as suggested in "thermal only" environment. P~~p~~rocure and accept industrial equipment as described above. Consider radiation effects as in "radiation only" DBE environment.~~Consider thermal capability as in "thermal only" DBE environment.~~
- If directly exposed to the HELB effects, ~~moisture,~~ a licensee should specify ~~specify~~ sealed components designed for high humidity or water spray conditions (e.g., ~~or~~ jungle-rated equipment). Procure and accept industrial equipment as described above. Consider radiation effects as in "radiation only" DBE environment. Consider thermal capability as in "thermal only" DBE environment.
- If existing data is insufficient to assure functionality, verify operability at the specified temperature and moisture conditions using industrial-type testing. Procure and accept industrial equipment as described above.

LOCA/MSLB—this type of environment typically occurs inside primary containment and involves exposure to a high pressure, high temperature steam/air mixture combined with high levels of radiation. Depending on the severity of these conditions the following options should be considered.

- Evaluate susceptibility of equipment to high pressure steam/air mixture and condensing moisture. If no seals are necessary, perform functional analysis.
- If seals are required, specify equipment with housings acceptable for pressure conditions and use electrical port seals. Perform functional analysis.

- Functional analysis—evaluate effects of high pressure, high temperature steam/air moisture on equipment functionality. Pressurization can have a direct effect on function (e.g., shift of I/P devices from DBE pressure). Evaluate moisture, radiation and temperature effects as suggested in the prior categories and by ~~thermal capability~~ using industry data.
- If existing data is insufficient to assure functionality, perform radiation, thermal or steam exposure. Procure and accept industrial equipment as described above.

~~Additional~~ Specific Examples

Identical Replacement Example #1

The licensee classified certain outside containment Limitorque motorized valve actuators as RISC-3. Some of these actuator applications were subject to LOCA conditions (radiation and increased ambient temperature) and/or HELB conditions (steam/air mixture with minimal pressure and radiation). The existing actuators were qualified by Limitorque reports. The licensee reviewed the qualification basis for the existing actuators and determined that the design and materials of construction for these actuators are identical to those used for commercial Limitorque SMB series actuators. Similar actuators are used in the plant's BOP applications. The licensee confirmed this information with the manufacturer and concluded that commercial SMB series actuators could be used for these applications.

Identical Replacement Example #2:

The licensee has several DC motor starters that were qualified to "DOR Guideline" criteria based on NSSS vendor qualification testing. For the starters that are now classified as RISC-3, the licensee will purchase identical replacement parts and units directly from the starter manufacturer.

Equivalent Replacement Example #1:

The licensee classified certain inside primary containment air-pilot solenoid operated valves as RISC-3. The existing valves were nuclear-grade styles (e.g. ASCO, NP series). The valves were normally energized and were de-energized to perform their design functions during LOCA and MSLB events. The licensee elects to procure from the same manufacturer commercial versions of the valves that have the same design characteristics but use some different materials. By selecting certain optional features the licensee is able to limit the differences to the coil and the use of nylon instead of stainless steel for one internal component. The scope of these differences was verified during discussions with the valve manufacturer. Since the valves were de-energized to function, a failure modes and effects analysis determined that coil failures would not prevent adequate operation. The licensee also identified a qualification report of a similar ASCO commercial valve with the nylon component that adequately functioned during equivalent LOCA/MSLB conditions. The licensee concluded that these differences were acceptable and that the commercial valves with certain options would function during the design service conditions. The procurement specification required the optional design features and certification of the procured valves to the specification.

Equivalent Replacement Example #2:

The Core Spray Pump Motor was classified as RISC-3 in this BWR. Motor refurbishment including an insulation system rewind was selected in lieu of motor replacement for this 400 horsepower motor located in the Reactor Building (outside primary containment). The design basis for the motor and its insulating system required functionality during the harsh environments associated with LOCA (no steam but increased ambient temperature and 10E6 rads gamma) and certain Reactor Building HELBs (180°F peak steam/air mixture temperature with minimal pressure or radiation). The licensee performed an evaluation of the proposed insulation system and concluded that there was reasonable assurance that the specified rewind insulation system would function under these conditions. He obtained a detailed list of the proposed insulating system materials from the commercial motor rewind shop and verified from published radiation data that they were all tolerant of the LOCA radiation levels. He also verified that during LOCA and post-LOCA operation the motor winding temperature would remain below the specified Class F thermal rating for the rewind system. Performance capability during HELB steam conditions was established by requiring the rewind insulating system to be designed, constructed, and tested as a commercial NEMA "sealed system". Such sealed systems demonstrate protection from external moisture by passing a NEMA-specified underwater high-potential test. He was also aware that motor-insulating systems with less protection had been successfully qualified to similar HELB conditions. In addition to other technical requirements, the procurement specification for the motor repair stipulated that the rewind must use the approved materials, was to be constructed as a NEMA sealed system, and must pass a NEMA sealed system high potential test. The vendor was required to provide certification to the technical procurement requirements and the sealed system test results.

New Equipment Design Example #1:

A post-accident monitoring pressure transmitter was located in the Auxiliary Building of this PWR and was required to function during the harsh steam conditions associated with certain HELBs in that building. The licensee elected to utilize a new design in lieu of identical replacement of the existing transmitter. Performance capability during HELB steam conditions was achieved by:

(1) selecting a commercial electronic transmitter whose maximum continuous operating temperature (150°F) was slightly lower that the calculated peak transient temperature during the worst case HELB,
(2) determining that during the HELB the heavy steel transmitter enclosure and cover provide sufficient thermal inertia to prevent the sensitive electronics from reaching the published maximum operating temperature, and
(3) utilizing a conduit seal that, in conjunction with the transmitter's cover o-ring seals, would prevent the external HELB steam/air conditions from penetrating into the transmitter.

The responsible engineer discussed the proposed design with the transmitter manufacturer who agreed that such a sealed transmitter should function during the postulated steam/air conditions. The manufacturer provided a copy of the industrial testing that demonstrated performance at the maximum operating temperature. In addition to other technical requirements, the transmitter procurement specification required certification to the manufacturer's published specification. The conduit seal was a grommeted commercial design that was rated for 500 psig and 350°F. The seal's procurement specification required certification to the manufacturer's published specification. The design change package specified the appropriate methods for installing the transmitter and seal including torquing requirements whenever the transmitter cover or seal were removed for maintenance or calibration.

New Equipment Design Example #2:

This licensee was experiencing operational problems with existing RISC-3 classified SRV discharge pressure switches and decided to replace them with a new design. The licensee selected a marine service explosion-proof pressure switch specially engineered for offshore applications and environmentally sealed against dust, water, oil and salt spray. The product literature indicated that seal integrity existed up to 250°F and the unit was rated for 10 Gs shock. The licensee contacted the manufacturer and was provided with the materials of construction. He verified that the internal materials and the lead wire potting (but not the lead wires) could tolerate the LOCA accident gamma dose. The LOCA/HELB temperature conditions were also slightly higher than the published continuous operation capabilities and the manufacturer had not verified sealing capability during combined high temperature and pressure conditions. Given the relatively short duration of temperatures above 250°F and material specifications for the potting compound, the manufacturer believed the units would remain function during LOCA/MSLB conditions. The licensee and the manufacturer agreed to modify the design to use a lead wire style that had been environmentally qualified per IEEE 323. They also agreed to conduct a high temperature, pressure steam test to verify functionality for the higher temperature portion (6 hours) of the LOCA/HELB. The procurement specification required the use of a particular wire type, the successful performance during the steam test, a copy of the test report, and certification to the procurement specification.

APPENDIX E

EVALUATION AND PROCUREMENT OF ITEMS IN RISC-3 APPLICATIONS PREVIOUSLY GOVERNED BY APPENDIX A TO 10 CFR 100

The application of Appendix A to 10 CFR Part 100, *Seismic and Geologic Siting Criteria for Nuclear Power Plants*, provides increased assurance, above that provided by industrial standards and processes, that the safety functions will be satisfied under design bases conditions. Prior to the issuance of §50.69, Appendix A to Part 100 requirements were applied to safety-related SSCs. Under §50.69, the safety significance of equipment is evaluated by applying a blend of risk-insights, operating experience and new technical insights. As a result, some equipment that is labeled as safety-related can be categorized as low safety-significant. As such, it is acceptable to apply standard industrial level standards and practices to equipment categorized as low safety-significant to assure that the design bases functions will be satisfied. These nationally recognized standards and established practices include methods for the seismic qualification of equipment and the determination of seismic design loads.

For low safety-significant SSCs, established industrial level practices and standards are used to design, procure and qualify RISC-3 equipment. Compliance with Appendix B to Part 50 and 10CFR Part 21 is not required because of the low safety-significance of RISC-3 SSCs. Alternate replacement equipment or equipment with new design may be procured from an industrial, non-Appendix B supplier. Condition monitoring and inspection under the preventative maintenance program provide additional assurance for issues such as two-over-one condition for equipment anchorage and spatial interaction.

Seismic design requirements should be considered in the procurement process. These design requirements are translated into a procurement specification through a direct incorporation of the design requirement or via a technical evaluation that is reflected in the procurement specification. The design criteria and methods to assure adequacy under seismic loading are based on established seismic design practices and industry standards, such as IBC 2000. Standard balance-of-plant receipt and inspection activities should be performed on equipment to ensure that the equipment received satisfies the procurement specifications.

The general processes used in the specification, procurement and evaluation of RISC-3 equipment can be grouped into one of the following options:

- Identical replacement
- Equivalent replacement
- Design change - new equipment or modifications

Identical Replacement

This refers to circumstances that involve identical design or configuration. No significant difference in the dynamic characteristics exists between the original and the replacement item. The suitability of design has been previously established and the seismic adequacy is maintained. No additional evaluation effort is required.

Equivalent Replacement

Equivalent replacement means that the design or configuration is not the same, yet it is not significantly different. The suitability of design has been established. The replacement equipment may not be dynamically similar to the original equipment. For such replacements, the seismic evaluation methods and acceptance criteria should be similar to those for safety-related, high safety-significant equipment. An evaluation should be performed to assess the effect that the design differences would have on the ability of the replacement item to perform its design function under the design basis seismic conditions. This evaluation may include a static or dynamic analysis (traditional or computer calculations), a review of test or seismic-experience data, a dynamic similarity or other qualitative engineering evaluation including documented engineering judgement, a review of operational experience, or any combination of these elements. Other available data sources for performing this evaluation include previously existing seismic qualification reports, EPRI reports and other valid industry guidance documents.

Design Change

A design change with new equipment design or modification should be used when:

- There are substantial design differences between the original and the replacement item,
- The item does not satisfy the equivalency determination,
- The item has been substantially modified or refurbished, or
- The item is completely new, i.e.: it does not replace an existing item.

The evaluation methods described for equivalent replacement may also be applicable to design changes installing new or modified equipment on a case-by-case basis. However, for more complex replacements and new designs where the equivalency evaluation methods are either not applicable or ineffective, a licensee may use an industrial consensus seismic standard that establishes the equipment seismic adequacy.

Consensus standards have been developed for industrial, non-nuclear applications in seismic areas. Among these are the National Earthquake Hazard Reduction Program (NEHRP), ASCE 7, Uniform Building Code (UBC), International Building Code (IBC), etc. These codes and standards have been used in the design of highly protected facilities and equipment. The latest revisions to these standards establish practices to determine the appropriate seismic loads and evaluation equipment functionality.

Option 2 does not change the [§50.2] design bases. Appendix A to 10 CFR Part 100 provides a detailed description of the seismic design bases. The Appendix A to Part 100 design bases are slightly different than that prescribed in the recent revision to the codes and standards. As such, an interim approach has been developed for use until the difference between the Part 100 design bases and the codes are reconciled.

General Approach – Application of 2000 International Building Code (IBC)

The IBC is one of the more recent codes and is supported by Building Association Code Administrators (BOCA), International Conference of Building Officials (ICBO) and Southern Building Code Congress International, Inc. (SBCCI). The current version of the code meets the intent of Appendix A to 10 CFR, Part 100 and explicitly addresses functionality as reflected by the use of a component importance factor, I_p, ranging from 1 to 1.5, in the calculation of seismic design force and the inclusion of equipment specific characteristics required to maintain functionality. For life-safety components required to function after an earthquake or for components containing hazardous material, the value of I_p is 1.5. This value of 1.5 should be assigned to applicable RISC-3 items of equipment when performing evaluations using the IBC.

The design basis input motions using the IBC could result in different seismic input motions at a specific location in a nuclear plant than those based on the plant's licensing basis in-structure response spectra. Thus, this interim approach follows the IBC with the exception of maintaining the design basis ground and in-structure response spectra and anchorage evaluation criteria. As a result, design basis floor seismic loads should be used in anchorage and structural load path evaluations.

Interim approach

Input Loads and Seismic Forces

The IBC describes maximum considered earthquake hazard ground motions for various regions, site-class definitions (rock, soil etc.) and mapped spectral response accelerations at different periods. It provides a procedure to develop a general design response spectrum curve and site specific procedure for determining ground motion acceleration. Using this information and the height in structure at the point of attachment of the component, equations are provided in the code to calculate seismic forces.

The seismic forces may be calculated using an alternative method. The parameters related to the IBC-based design spectral response acceleration and height in structure in the calculation of seismic force are substituted by plant-specific licensing basis spectral parameters at the building and location where the equipment is mounted. The IBC equations take into account a component amplification factor (varying from 1.0 to 2.5) and a component response modification factor (varying from 1.0 to 5.0), which are used in accordance with the code.

Equipment Anchorage

In performing equipment anchorage calculations, plant's licensing basis in-structure response spectra are used as seismic input rather than the spectral parameters defined in IBC. The acceptance criteria, factors of safety etc. for anchorage calculation are also in accordance with the plant's licensing basis for safety-related equipment.

Equipment Load Path, Attachments and Supports

For internal load path analyses where necessary, and for the analyses of attachments to equipment support, the plant's licensing basis in-structure response spectra are used as seismic input. In addition to the calculation of lateral forces, other guidance and criteria from the code, including consideration of specific equipment caveats provided in the code should be followed. The interim approach utilizes equations for calculating seismic forces and other guidance and criteria from the IBC, whereas the input loads (i.e., in-structure response spectra) are based on the plant's licensing basis.

The approach uses a combination of elements from the IBC (most recent national consensus standard) and the plant's seismic design basis. A comparative review should be performed of the design response spectra and in-structure amplified response spectra parameters in the IBC code at regions in the US with the highest seismic ground response spectra and in-structure amplified response spectra. Such a review assures that the use of IBC code's equations, functionality caveats and other criteria are consistent and appropriate when combined with the input loads and spectra from sources outside of the IBC code (i.e., the plant's design basis).

Table E-1 provides a summary of the interim approach.

General Pilot Plant Example

For one of the lead/pilot plants, the ground and in-structure response spectra at these high seismic regions from the IBC were compared to the plant specific design bases spectra. The spectra from the IBC at these high seismic regions are significantly higher than the corresponding licensing basis ground and in-structure response spectra for the lead nuclear plant in the entire frequency range. Based on the margins in this comparison, a similar conclusion is expected for other US nuclear plants. Since the IBC equations and other guidance and criteria are applicable to all US regions, including regions with the highest spectra, it follows that the use of seismic input or in-structure response spectra from the plant's licensing basis will not invalidate the code criteria or guidance.

Table E-1
Summary of Interim Seismic Approach for RISC-3 SSCs

Elements of Seismic Review Process	**Interim RISC 3 Seismic Approach**
Anchorage	Design Basis Loads and Allowables; IBC Caveats (e.g., no friction clips)
Seismic Forces and Displacements	Higher of Design Basis and IBC Loads; IBC Equations and Criteria*
Equipment Critical Structural Components/ Load Path	Higher of Design Basis and IBC Loads; IBC Caveats, Criteria & Allowables*
Equipment Attachments and Supports	Higher of Design Basis and IBC Loads; IBC Caveats, Criteria & Allowables* (e.g., externally attached items, cable trays etc.)
Functionality	IBC Caveats & Restrictions**
Interaction	Higher of Design Basis and IBC Loads; IBC Caveats & Criteria*

*IBC equations are slightly modified to eliminate duplication of amplification of ground spectra at a specific height in the structure, if design-basis in-structure response spectra (ISRS) are used, since any amplification through the structure is addressed by them.
** Use of a component importance factor I_p of 1.5 for all RISC-3 items of equipment confirms functionality following a seismic event.

Specific Example

Later