NUCLEAR ENERGY INSTITUTE

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June 28, 2002

Mr. David B. Matthews
Director, Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Matthews:

Enclosed for NRC staff review is Draft Revision C to NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline.* The document has been completely rewritten to reflect the NRC observations and comments provided in your February 8, 2002 letter, the NRC-industry interactions that have taken place since NEI 00-04, Rev. B was submitted in June 2001, and the lessons learned from the pilot plant SSC categorization activities.

The two most significant changes are:

- The document now focuses solely on SSC categorization, with treatment being addressed in a separate internal industry Option 2 Technical Basis Document, and
- The categorization process now identifies safety-significant functions and maps structures, systems and components (SSCs) to the safety-significant functions with a comparison against SSC risk significance derived from the PRA.

The guideline retains the integrated decision-making process for categorizing SSCs using a multi-disciplined integrated decision-making panel of experienced licensee designated personnel to oversee and categorize SSCs.

Enclosure 1 is the revised guideline. Enclosure 2 is summary of the major changes. We will send you a detailed summary of the industry's disposition of NRC comments on NEI 00-04, Rev. B early next week.

Y601
DDD: D Matthews to eRIDS

We look forward to discussing the revised guideline and the pilot project at our scheduled July 10, 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

Anthony R. Pietrangelo

Enclosures

NEI 00-04 (DRAFT - Revision C)

10 CFR 50.69 SSC Categorization Guideline

NUCLEAR ENERGY INSTITUTE

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This report has been prepared by the NEI Risk Applications Task Force, the NEI Option 2 Task Force, and the NEI Risk-Informed Regulatory Working Group

NOTICE

TABLE OF CONTENTS

1 INTRODUCTION

This document provides detailed guidance on categorizing structures, systems and components for licensees that choose to adopt 10 CFR 50.69, *Scope of Structures, Systems and Components, Governed by Special Treatment Requirements*. A licensee wishing to implement §50.69 makes a submittal, consistent with the example described in Appendix B of this guideline, to the Director of Nuclear Reactor Regulation, NRC for review and approval. Licensees that commit to implementing §50.69 in accordance with this guideline should expect minimal NRC review.

This guidance is based on the principles of NRC Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, namely:

1. The initiative should result in changes that are consistent with defense-in-depth philosophy.
2. The initiative should result in changes that maintain sufficient safety margins.
3. Performance measurement strategies are used to monitor the change.
4. The implementation of the §50.69 initiative should not result in more than a minimal increase in risk.
5. The risk should be consistent with the Commission's safety goal policy statement.

There are two segments associated with the implementation of 10 CFR 50.69: the categorization of structures, systems and components; and the application of NRC special treatment requirements[1] consistent with the safety significance of the equipment categorized in the first step. This guidance deals the alternative categorization of structures, systems, and components per §50.69. The application of special treatment regulations and controls is a function of the SSC categorization. The existing special treatment provisions for RISC-1 and RISC-2 SSCs are maintained or enhanced to provide reasonable assurance that the safety-significant functions, identified in the §50.69 process will be satisfied. RISC-3 and RISC-4 SSCs are governed by the treatment requirements, described in 10 CFR 50.69.

1.1 BACKGROUND

The regulations for design and operation of US nuclear plants define a specific set of design bases events that the plants must be designed to withstand. This is known as a deterministic regulatory basis because there is little explicit consideration of the probability of occurrence of the design basis events. It is "determined" they will occur,

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

and the plant is designed and operated to prevent and mitigate such events. 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 events 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 the deterministic regulatory basis, over 2700 reactor years of operation have been accumulated in the US (over 10,000 reactor years worldwide), with a corresponding body of data relative to actual transients, accidents, and plant equipment performance. Such data is used in modeling accident sequences (including sequences not considered in the deterministic regulatory basis) to estimate the overall risk from plant operation. Further, each US plant has performed a probabilistic risk analysis (PRA), which uses these data. PRAs describe risk in terms of the frequency of reactor core damage and 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). Also, the NRC has provided guidance in Regulatory Guide 1.174, on how to use risk-insights to change the licensing basis.

In 1999, the Commission approved a NRC staff recommendation to expand the scope of risk-informed regulatory reforms. The Commission directed the NRC staff to develop a series of rulemakings that would provide licensees with an alternative set of requirements in two areas: NRC technical requirements, and requirements that define the scope of structures, systems and components (SSCs) that are governed by NRC special treatment requirements.

1.2 REGULATORY INITIATIVE TO REFORM THE SCOPE OF EQUIPMENT AND ACTIVITIES SUBJECT TO NRC SPECIAL TREATMENT REQUIREMENTS

The objective of this regulatory initiative is to adjust the scope of equipment subject to special regulatory treatment (controls) to better focus licensee and NRC attention and resources on equipment that has safety significance. This guideline addresses the use of risk insights to define the scope of equipment that should be subject to NRC special treatment provisions as defined in §50.69.

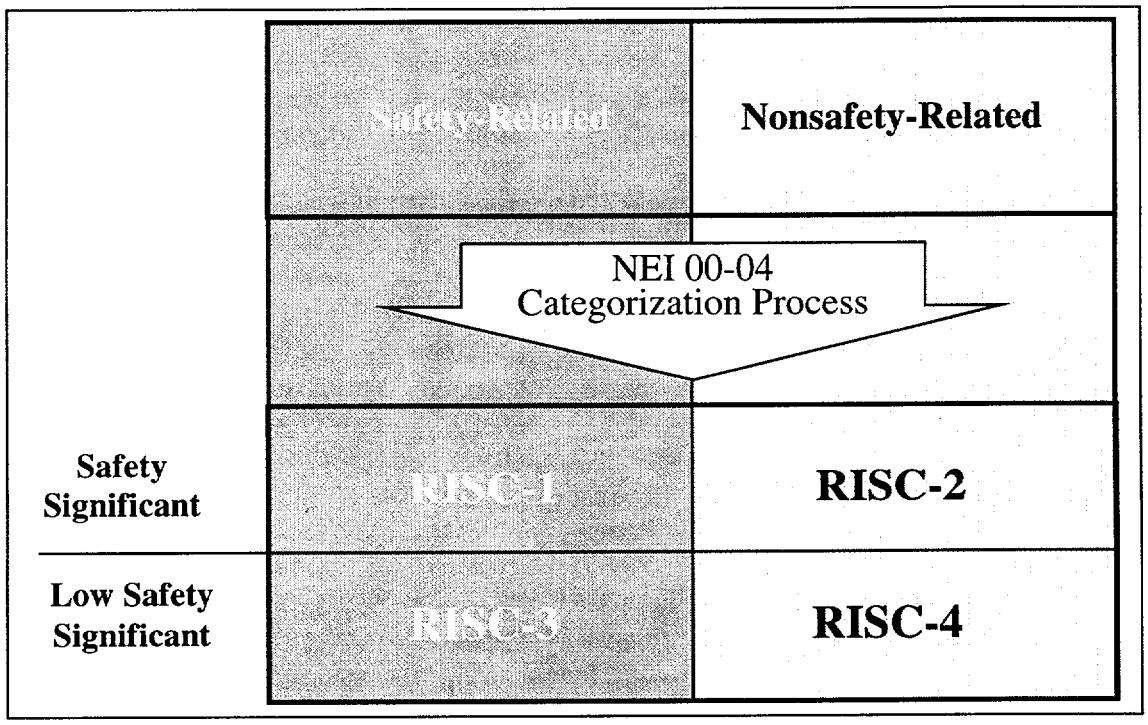
Current NRC regulations define the plant equipment necessary to meet the deterministic regulatory basis as "safety-related." This equipment is subject to NRC special treatment regulations. Other plant equipment is categorized as "nonsafety-related", and is not subject to special treatment requirements. There is a set of nonsafety-related equipment that is subject to a select number of special treatment requirements or a subset of those requirements. This third set is often referred to as "important-to-safety." Generally,

licensees apply augment quality controls (a subset of the criteria in Appendix B to Part 50) to these "important to safety" SSCs.

§50.69 does not replace the existing "safety-related" and "non safety-related" categorizations. Rather, §50.69 divides these categorizations into two subcategories based on high or low safety significance. The §50.69 categorization scheme is depicted in Figure 1-1, and detailed guidance is provided in Sections 2 through 10.

The §50.69 SSC categorization process is an integrated decision-making process. This process blends risk insights, new technical information and operational feedback through the involvement of a group of experienced licensee-designated professionals. This group, known as the Integrated Decision-Making Panel (IDP), is supported by additional working level groups of licensee-designated personnel, as determined by the licensee.

Figure 1-1
RISK INFORMED SAFETY CLASSIFICATIONS (RISC)



The §50.69 categorization process will identify some safety-related SSCs as being of low or no safety-significance and these will be recategorized as RISC-3 SSCs, while other safety-related SSCs will be identified as safety-significant, and be recategorized as RISC-1. Likewise, some nonsafety-related SSCs will be recategorized as safety-significant (RISC-2) and others will remain of low or no safety-significance, and be recategorized as

RISC-4 SSCs. For the purposes of implementing §50.69, "important to safety" SSCs enter into the categorization process as "non safety-related." Thus, safety-related SSCs can only be categorized as RISC-1 or RISC 3, and nonsafety-related SSCs, including the "important to safety" SSCs can only be categorized as RISC-2 or RISC-4.

Those SSCs that a licensee chooses not to evaluate using the §50.69 SSC categorization process remain as safety-related, nonsafety-related and "important to safety" SSCs.

1.3 GUIDING PRINCIPLES

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

- Use applicable risk assessment information.
- Deterministic or qualitative information should be used, if no PRA information exists related to a particular hazard or operating mode.
- The categorization process should employ a blended approach considering both quantitative PRA information and qualitative information.
- The Reg. Guide 1.174 principles of the risk-informed approach to regulations should be maintained.
- A SSC retains its original categorization if a basis for re-categorization cannot be developed.
- Attribute(s) that make a SSC safety-significant should be documented.

1.4 VOLUNTARY AND SELECTIVE IMPLEMENTATION

US nuclear generating plants have attained and maintained outstanding safety performance record. The existing NRC regulations together with the NRC's regulatory oversight and inspection processes clearly provide adequate protection of public health and safety. As a result, the decision to adjust and improve the scope of equipment that is subject to NRC special treatment requirements is a voluntary, licensee decision. Each licensee should make its determination to adopt the new rule 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, and hence an overall improvement in safety.

From previous risk-informed activities, a licensee is already aware of the areas where the §50.69 categorization process would provide a benefit. As a result, a licensee can determine the appropriate set of equipment to recategorize under §50.69, and schedule the implementation over a period of time. The SSC categorization schedule should be sent to the NRC as part of the licensee's implementation submittal (see Appendix B).

2 OVERVIEW OF CATEGORIZATION PROCESS

The overall process used in categorizing SSCs for the purposes of changing the special treatment requirements under 10CFR50.69 is depicted in Figure 2-1. This process builds upon the insights and methods from many previous categorization efforts, including risk-informed IST and risk-informed ISI. It is intended to be a comprehensive, robust process that includes consideration of various contributors to plant risk and defense-in-depth.

The process includes eight primary steps:

- Assembly of Plant-Specific Inputs
- System Engineering Assessment
- Component Safety Significance Assessment
- Defense-In-Depth Assessment
- Preliminary Engineering Categorization of Functions
- Risk Sensitivity Study
- IDP Review and Approval
- SSC Categorization

Each of these steps is covered in more detail in subsequent section of this document. This section provides a brief overview of the elements of each step and the inter-relationships between steps.

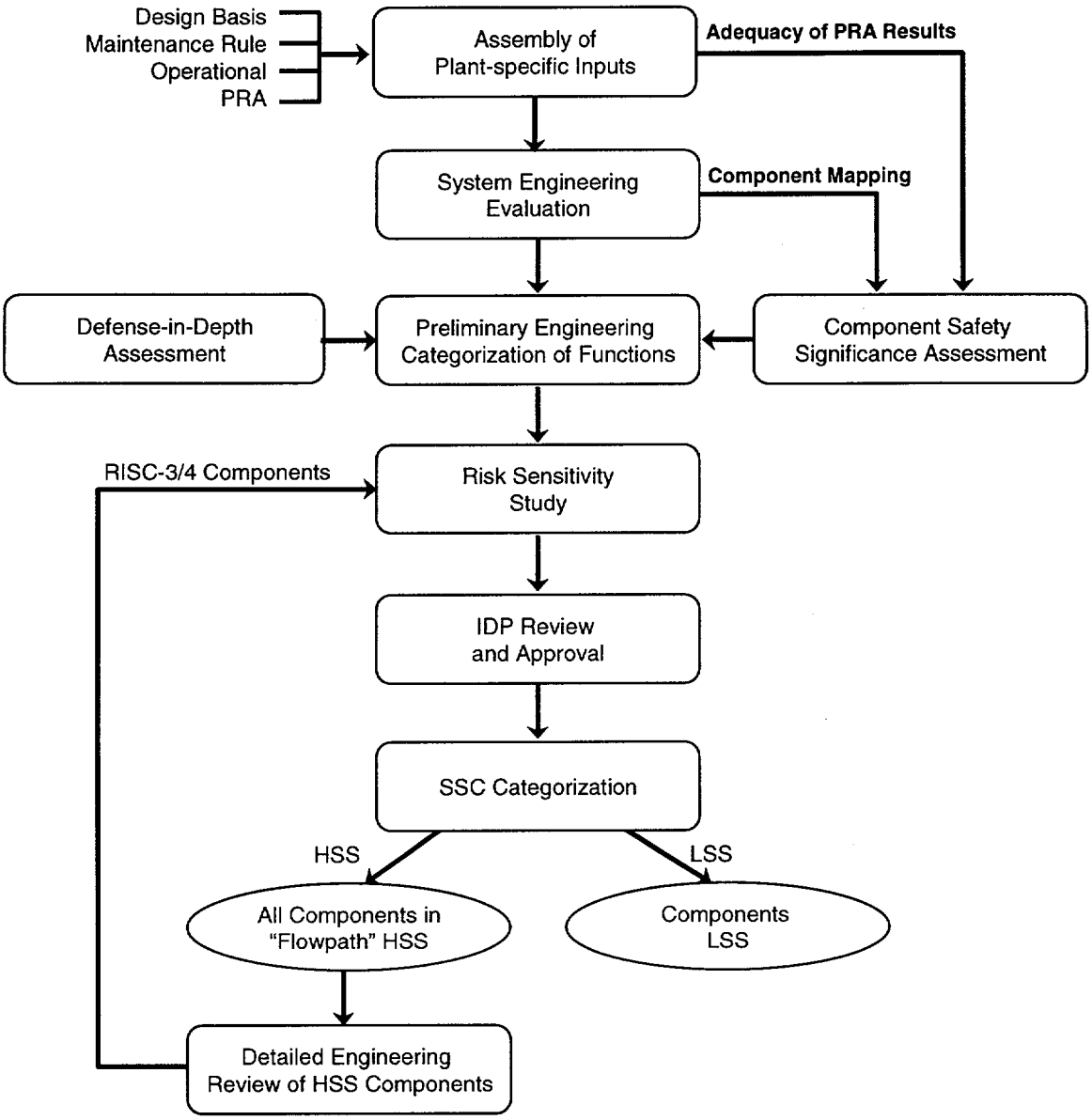
Assembly of Plant-Specific Inputs

This step involves the collection and assessment of the key inputs to the risk-informed categorization process. This includes design and licensing information, PRA analyses, and other relevant plant data sources. In addition, this step includes the critical evaluation of plant-specific PRA analyses to assure that they are adequate to support this application. More detail is provided on this step in Section 3.

System Engineering Assessment

This task involves the initial engineering evaluation of a selected system to support the categorization process. This includes the definition of the system boundary to be used and the components to be evaluated, the identification of system functions, and a coarse mapping of components to functions. The system functions are identified from a variety of sources including design/licensing basis analyses and PRA analyses. The mapping of components is performed to allow the correlation of PRA importance measures to system functions. More detail is provided on this step in Section 4.

Figure 2-1
RISK-INFORMED CATEGORIZATION PROCESS



Component Safety Significance Assessment

This step involves the use of the plant-specific PRA analyses to identify components that are to be considered safety significant. The process includes consideration of the component contribution to full power internal events risk, fire risk, seismic risk and other external hazard risks, as well as shutdown safety. More detail is provided on this step in Section 5.

Defense-In-Depth Assessment

This step involves the evaluation of the role of components in preserving defense-in-depth related to core damage, large early release and long term containment integrity. More detail is provided on this step in Section 6.

Preliminary Engineering Categorization of Functions

This step involves integrating the results of the two previous tasks to provide a preliminary categorization of the safety significance of system functions. This includes consideration of both the risk insights and defense-in-depth assessments. More detail is provided on this step in Section 7.

Risk Sensitivity Study

The preliminary categorization is used to identify the SSCs that may be low safety significant. A risk sensitivity study is performed to investigate the aggregate impact of potentially changing treatment of those low safety significant SSCs. More detail is provided on this step in Section 8.

IDP Review and Approval

The Integrated Decision-Making Panel (IDP) is a multi-disciplined team that reviews the information developed by the categorization team. The Integrated Decision-making Panel (IDP) uses the information and insights developed in the preliminary categorization process and combines that with other information from design bases and defense-in-depth to finalize the categorization of functions. More detail is provided on this step in Section 9.

SSC Categorization

When the IDP approves the categorization of system functions, then the initial coarse mapping of components to system function may be used to define the safety significant SSCs. Additionally, the licensee may elect to perform a more detailed evaluation of the system and components that have been categorized as safety-significant to identify those SSCs that can be categorized as low safety-significant because a failure of these SSCs would not inhibit a safety-significant function. In the event this more detailed review identifies any new low safety significant SSCs, the results of that re-categorization is reevaluated in the risk sensitivity study and provided to the IDP for final review and approval. More detail is provided on this step in Section 10.

3 ASSEMBLY OF PLANT-SPECIFIC INPUTS

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.

3.1 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:

- Full Power 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 designation)
- UFSAR
- Design Basis Documents
- 10 CFR 50.2 Assessments
- 10 CFR 50.65 information

3.2 Use of PRA Information

An essential element of the SSC categorization process is a plant specific PRA model of the internal initiating events at full power operations. The PRA should be of a standard that satisfies the accepted standards for PRA technical adequacy, reflect the as-built and as-operated plant, and quantify core damage frequency (CDF) and large early release frequency (LERF) for power operations due to internal events. Assessments of other hazards and modes of plant operation should be reviewed to 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 broader scope PRAs are available.

The PRA should be consistent with accepted practices, in terms of the scope and level of detail for the hazards evaluated. PRA adequacy can be assured through a peer review of the PRA, as described in NEI 00-02 (Ref. 9) as amended to incorporate NRC comments provided in NRC letter to NEI dated April 2, 2002 (Ref. 15). Following the guidance in NEI 00-02 help ensure appropriate scope, level of detail, and quality of the PRA. The

ASME PRA Standard (Ref. 17) provides a consensus process for defining the attributes of a PRA that are necessary to support an application like the categorization process. When available, the other industry consensus standards on PRA are also an acceptable means to assure acceptability of the PRA results. Where available, industry processes for using a combination of the peer review process and standards should be utilized to maximize the benefit of both processes.

The licensee should ensure that documentation exists for the review process, the qualification of the reviewers, the summarized review findings, and resolutions to these findings. Based on the PRA peer review 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 process is a series of grades in a spectrum of technical areas. Areas with low grades (grades less than 3) should be reviewed and evaluated to assess whether changes in the PRA are necessary.

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 measures. The following describes methods acceptable to ensure that the PRA is of sufficient quality to be used for regulatory decisions and meets the quality standards described in Reg. Guide 1.174, and includes measures such as:

- 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 program can be used as an important element in this process).
- Provide documentation and maintain records in accordance with licensee practices.
- Provide for an independent review of PRA completeness (an independent peer review 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 used to support the categorization process, provided it can be shown that the appropriate quality provisions have been met.

The non-PRA aspects of the categorization process should be subject to the normal licensee quality assurance practices, including the applicable provisions of the licensee's Appendix B quality program for safety-related SSCs.

3.3 Characterization of the Adequacy of PRA Results

Figure 3-1 depicts the approach to be employed in ensuring the adequacy of PRA information used in the categorization of SSCs. This process is consistent with the approach proposed by the industry for making use of industry peer reviews in demonstrating that the ASME PRA Standard has been met. It is anticipated that the Regulatory Guide under development on assessing the adequacy of PRAs will be similar to this approach also. This new regulatory guide will establish the common process for demonstrating that the results from a plant-specific PRA are adequate for the application being undertaken.

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 ensuring the adequacy of the base internal events PRA results. The NEI 00-02 peer review provides several outputs, which are useful in characterizing the PRA results. The first output is a set of element grades, ranging from 1 to 4, which provide 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 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 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.

The second important output of the NEI 00-02 peer review process are the Fact and Observations (F&Os) that document the strengths and weaknesses of specific aspects of the PRA. F&Os 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 ensure that (1) none of the internal event peer review 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 the adequacy of the PRA. This

characterization should be provided to the IDP as a basis for the adequacy of the PRA information used in the categorization process and should be summarized in the submittal to the NRC. At a minimum, this characterization should include the following:

Internal Events PRA (Full Power 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 of the internal events PRA including elements 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.
- Considerations identified by the NRC in their letter to NEI [Ref. 15].

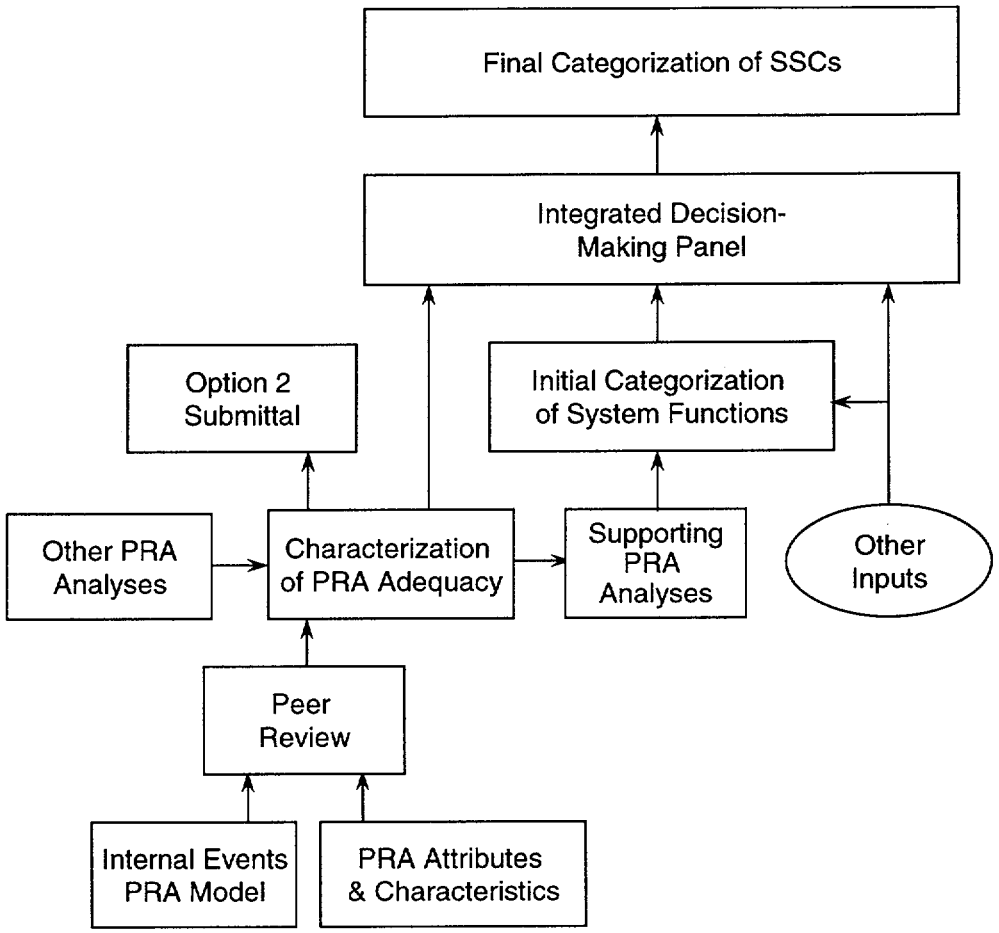
Other PRA Analyses

- A basis for why the other PRA analyses adequately reflect the as-built, as-operated plant.
- A disposition of the impact of elements 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) should use this information, in combination with the results of the categorization analyses and other information, to recommend a categorization for each function/SSC. The process to be used to justify the adequacy of the PRA information is also summarized in the submittal to the NRC.

Figure 3-1

PROCESS FOR ASSURING PRA ADEQUACY FOR OPTION 2 CATEGORIZATION



4 SYSTEM ENGINEERING ASSESSMENT

The system engineering assessment involves the identification and development of the base information necessary to perform the risk-informed categorization. In general, it includes the following elements:

- System Selection and System Boundary Definition
- Identification of System Functions
- Coarse Mapping of Components to Functions

System Selection and System Boundary Definition

This step includes defining system boundaries where the system interfaces with other systems. The bases for the boundaries can be the equipment tag designators or some other means as documented by the licensee. All components and equipment of the chosen system should be included. However, care should be taken in extending beyond system boundaries to avoid the introduction of new systems and functions. For example, many systems require support from other systems such as electric power and cooling water. The system boundary should be defined such that any components from another system, only support the safety function of the primary system of interest. This may lead to the inclusion of some power breakers in the system boundary, but would probably exclude the MCC or bus.

Identification of System Functions

This step involves the identification of all system functions. A variety of sources are available for the identification of unique system functions including:

- Design Basis Safety Functions
- Maintenance Rule Functions
- Functions Modeled in the Plant-specific PRA
- Operational Functions

All design basis functions and beyond design basis functions identified in the PRA should be used. The system functions should be consistent with both the functions defined in the design basis documentation and the maintenance rule functions. While beyond design basis functions may be included in the maintenance rule functions, a review of the PRA should be conducted to assure that any function for the chosen system that is modeled in the PRA is represented. The system function should also be reviewed to assure that any special considerations for external events, plant startup / shutdown and refueling are also represented. Some functions may be further subdivided to allow discrimination between potentially safety significant and low safety significant functions associated with a flow path.

Coarse Mapping of Components to Functions

This step involves the initial breakdown of system components into the system functions they support. System components and equipment associated with each safety-significant function are identified and documented. There are several options to this implementation element:

1) Define the flow path associated with each function and then define the components associated with that function. In this case, the flow path definition must consider branch lines and interfaces with other flow paths to assure that the entire flow path is appropriately modeled and the boundaries clearly delineated.

2) If passive components have been categorized according to guidance for risk-informed ISI, the risk-informed segments are a good starting point. There would be additional benefit, if the SSC categorization for passive components using the ASME Code Case N-658, *Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities* (Ref. 16), is being implemented at the same time.

   In these cases, for each of the system functions from the previous step, the ISI segments associated with that function must be defined. That is, the flow path for each function is defined in terms of ISI segments. If the SSCs associated with an ISI segment have already been defined in the risk-informed ISI program, the only additional work is:
   a. Associate piece parts with a component that has already been categorized in the ISI program and,
   b. Create new equivalent ISI segments for portions of the system that may not have been in the scope of the RI ISI program.

   This is conservative because not every component in an ISI segment for each function is required to support that function.

Note that for either alternative, some functions (e.g., instrumentation to support the function, or isolation of the function) have no true flow path, but the components associated with these functions can be readily identified from system drawings once the system boundaries are identified.

Although this step involves the assignment of SSCs to a given flow path, this is not the primary focus of this step. In a later subsequent step, the categorization of the flow paths represented by each function will be presented to the IDP for review. The assignment of SSCs to the flow paths representing each of the functions is necessary at this step to ensure that every SSC with a tag identifier for the system being considered is represented in at least one of the functions. If SSCs are identified that are not assigned to at least one function, then new function(s) should be created for those SSCs.

5 COMPONENT SAFETY SIGNIFICANCE ASSESSMENT

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 5-1.

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.

Importance from Internal Events

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 system functions is categorized 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:

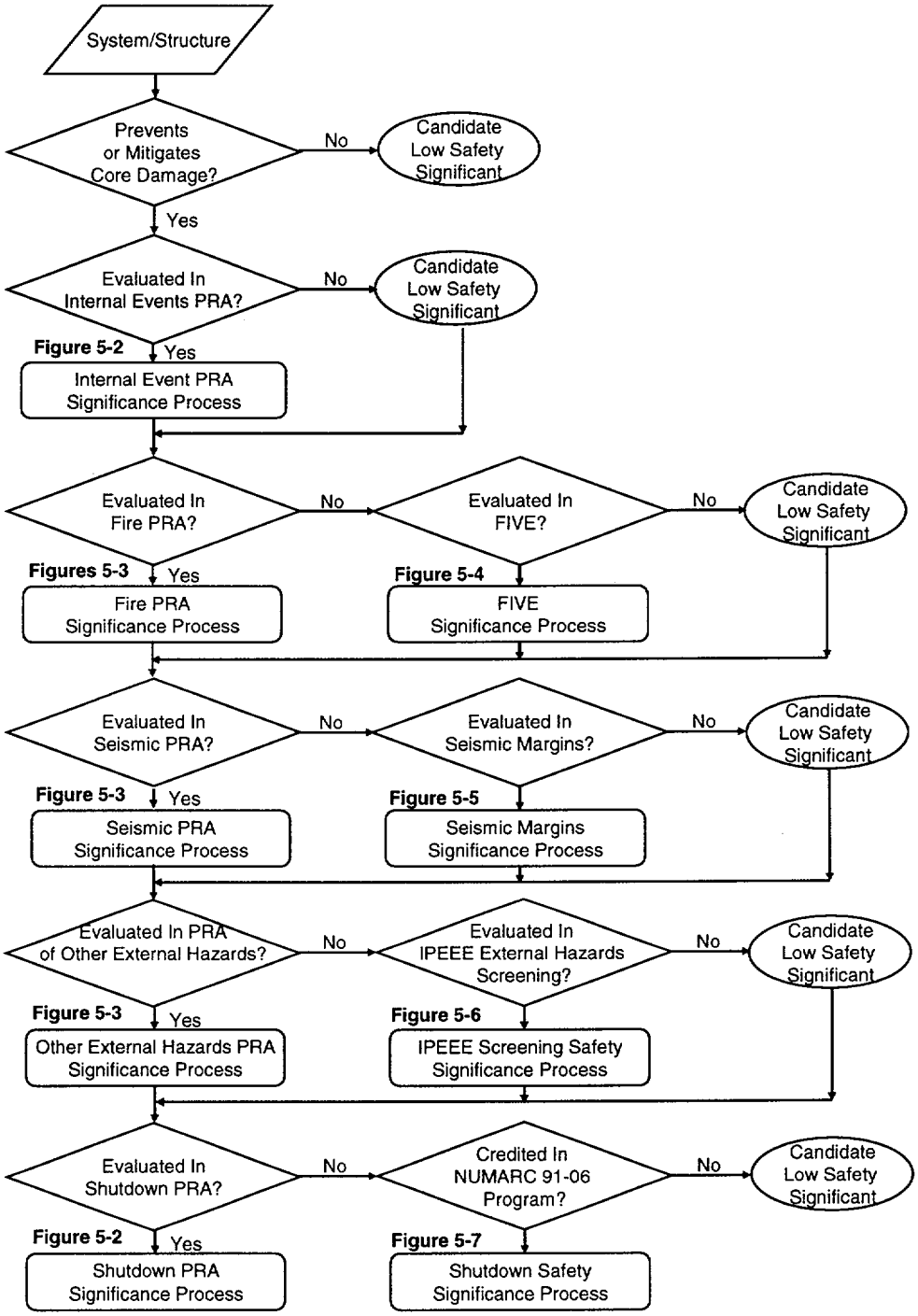
- Can it produce a potential initiating event?
- Is it credited for prevention of core damage or large early release?
- Is it necessary for another system or structure evaluated in the PRA to prevent an event or mitigate an event?

Some systems and structures are implicitly modeled in the PRA. It is important that PRA personnel that are knowledgeable in the scope, level of detail, and assumptions of the plant specific PRA make these determinations. As outlined in Section 1, by focusing on the significance of system functions and then correlating those functions to specific components that support the function, it is possible to address even implicitly modeled components. 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 5.1.

If the system/structure is not evaluated in the internal events PRA, then the assessment of the safety categorization relative to internal events is performed and then reviewed and approved by the IDP to determine. In either case, the evaluation is continued with fire risk.

Figure 5-1

USE OF RISK ANALYSES FOR SSC CATEGORIZATION



Importance from Fire Events

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. In making this determination specific attention should be given to structures and the role the play as fire barriers in the fire PRA. It is important that PRA personnel that are knowledgeable in the scope, level of detail, and assumptions of the plant specific fire PRA make the determinations with respect to fire PRAs. 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 5.2.

If the plant does not have a fire PRA, a fire risk evaluation is required, such as the *EPRI Fire Induced Vulnerability Evaluation (FIVE).* Again, it is important that personnel that are knowledgeable in the scope, level of detail, and assumptions of the fire risk evaluation (FIVE) make these determinations. 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 5.2.

If the system/structure is not involved in either a fire PRA or FIVE evaluations, then the assessment of the safety classification relative to fire risks is performed and then reviewed and approved by the IDP.

Importance from Seismic Events

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. Again, it is important that PRA personnel that are knowledgeable in the scope, level of detail, and assumptions of the plant specific seismic PRA make these determinations. 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 5.3.

If the plant does not have a seismic PRA, then a seismic risk evaluation, such as a seismic margin evaluation that was performed in response to the IPEEE should be performed. The seismic importance should be determined by personnel knowledgeable in the scope, level of detail, and assumptions of the seismic margins analysis. If the system or structure is included 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 5.3.

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 performed and then reviewed and approved by the IDP.

Importance from Other External Events

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 should make these determinations. 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 5.4.

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 should make these determinations. If the system or structure is 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 5.4.

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 performed and then reviewed and approved by the IDP.

Importance from Shutdown Events

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 should make the determination. If the system or structure is 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 5.5.

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 should 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 5.5.

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 performed and then reviewed and approved by the IDP.

5.1 Internal Event Assessment

The significance of SSCs that are included in the internal events PRA is evaluated using Figure 5-2. 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 used to address the initiating event role of the SSC. The mitigation importance of the SSC is assessed using the available importance measures.

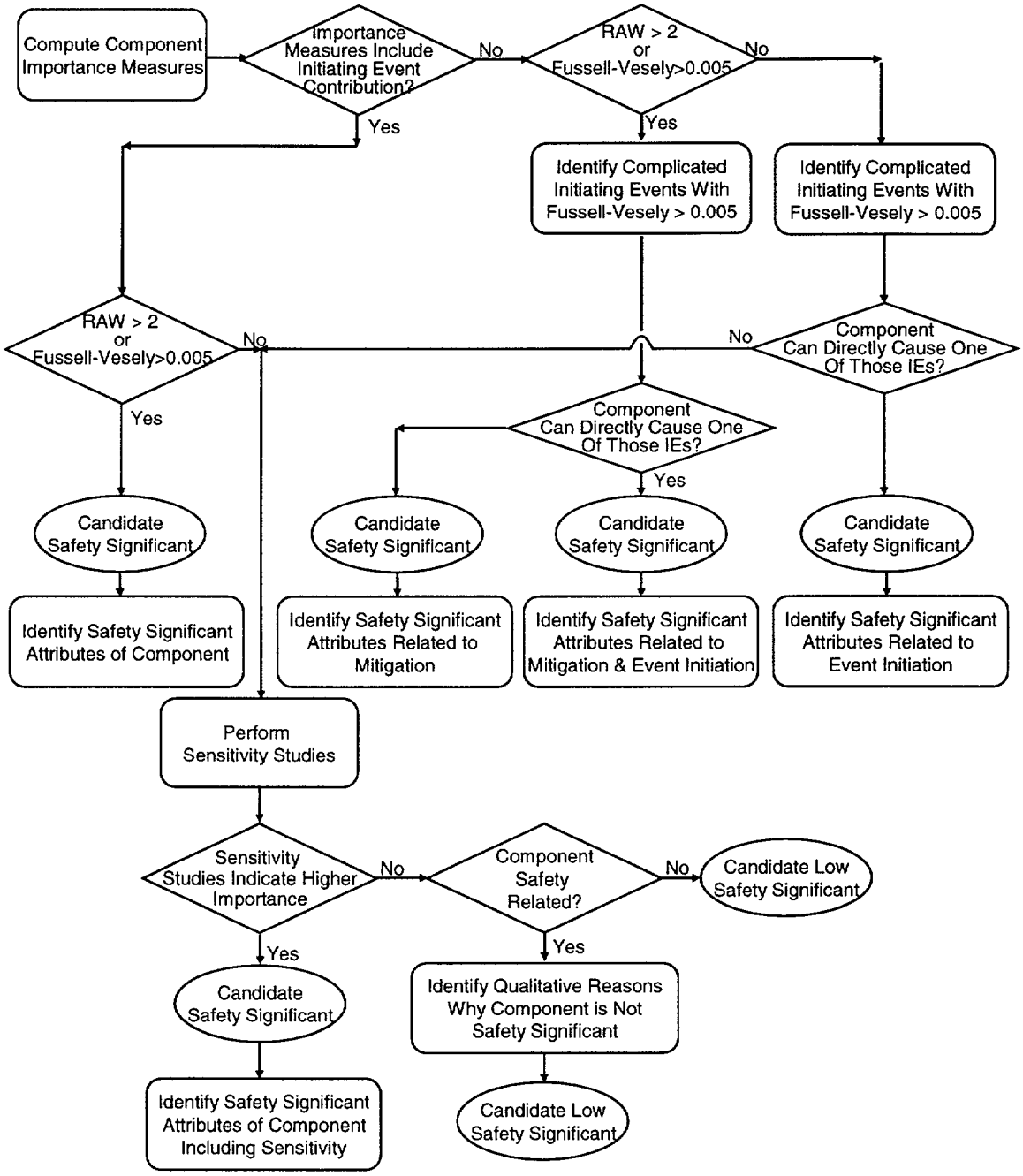
The qualitative process questions whether the SSC can directly cause a complicated initiating event that has a Fussell-Vesely importance greater than the criteria (0.005). If it does, then it is considered a candidate safety significant SSC and the attributes that could influence that role as an initiating 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 that represent the SSC. This can include events that explicitly model the performance of an SSC (e.g., pump X fails to start), events 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 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 [do we need examples]. For obtaining risk insights from the PRA for passive pressure boundary components, additional guidance is provided in ASME Code Case N-658, Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities. Guidance for categorization (and special treatment) for in-service 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 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 that are expected to be affected by the special treatment requirements being evaluated.

Figure 5-2

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



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. The RAW for common 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):

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.

Table 5-1
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 were not met. The component failure 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 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 categorization should be identified to the IDP. In many cases, special treatment should have little or no impact on such SSCs. If the IDP determines that this is the case, it may decide to categorize 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 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 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 5-2.

Table 5-2
Sensitivity Studies For Internal Events 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 • Set all maintenance unavailability terms to 0.0 • Any applicable sensitivity studies identified in the characterization of PRA adequacy

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. In cases where plant-specific uncertainty distributions are not readily available, other PRAs should be reviewed to identify appropriate parameter ranges. Experience with plant-specific PRAs has shown that the variations in distributions are relatively small, especially with respect the ratio of the mean and 95th 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 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, it is a candidate for RISC-3. In this case the analyst is

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) 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.

5.2 Fire Assessment

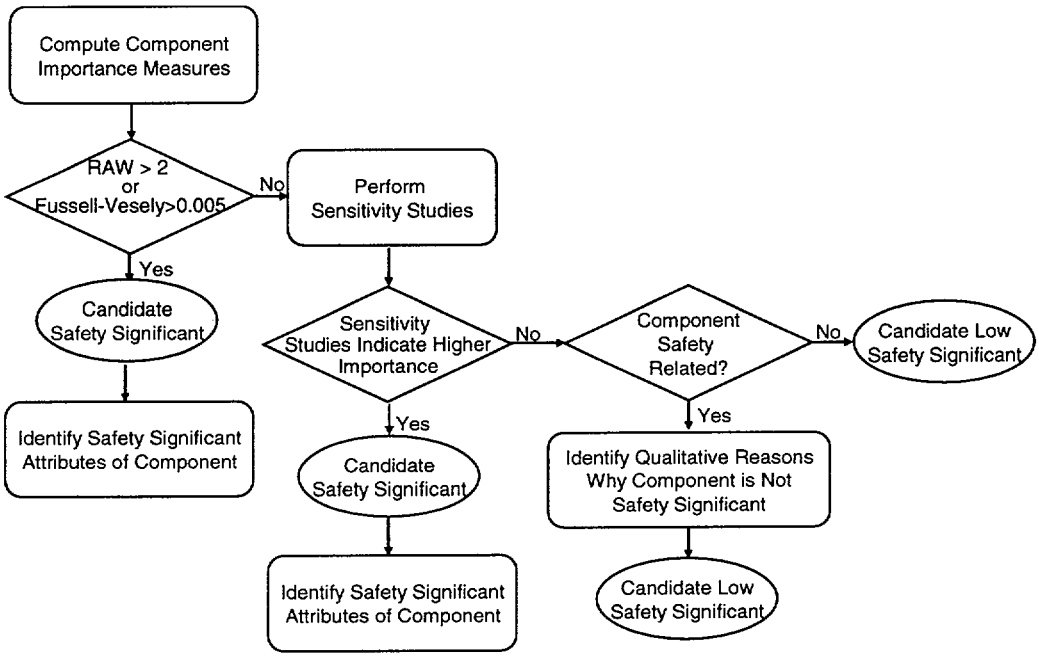
The fire safety significance process takes one of two forms. For plants with a fire PRA, the process is similar to that described for an internal events PRA. This process is shown on Figure 5-3, and is discussed below. Plants that relied upon a FIVE analysis to assess fire risks for the IPEEE should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

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. Aside from that small change, the process is the same as the internal events PRA process.

Fire suppression systems that are evaluated using the fire risk analysis can be categorized using this process. However, in order to apply this categorization process to suppression systems, specific sensitivity studies may be required to identify their relative importance, consistent with Fussell-Vesely and RAW (guarantee success/failure). 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. Once again, the use of sensitivity studies can be beneficial in identifying the role a barrier plays in maintaining risk levels.

Figure 5-3

RISK IMPORTANCE PROCESS FOR COMPONENTS ADDRESSED IN FIRE, SEISMIC & OTHER EXTERNAL HAZARD PRAs



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.

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, 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.

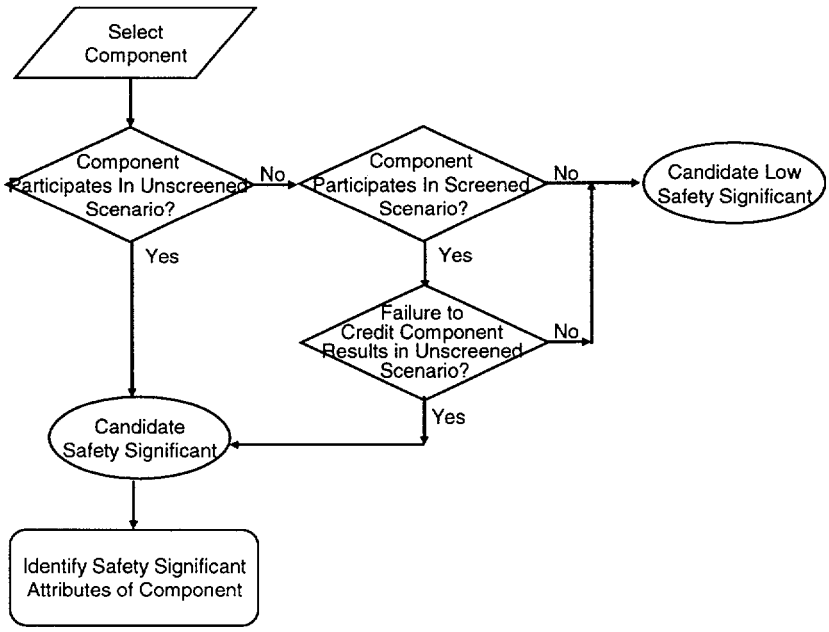
The recommended sensitivity studies for fire PRA are identified in Table 5-3.

Table 5-3
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 • Set all maintenance unavailability terms to 0.0 • All manual suppression =1.0 • Any applicable sensitivity studies identified in the characterization of PRA adequacy

The FIVE methodology is a screening approach to evaluating fire hazards. It does not generate numbers, 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 5-4.

Figure 5-4
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN FIVE



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 safety significant 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 a candidate for the low safety significant category.

5.3 Seismic Assessment

The seismic safety significance process takes one of two forms. For plants with a seismic PRA, the process is similar to that described for a fire PRA. This process is shown on Figure 5-3 and discussed below. Plants that relied upon a seismic margins analysis to assess seismic risks for the IPEEE would use the modified process shown in Figure 5-5.

The generalized safety significance process for plants with a seismic PRA is the same as the process for a fire 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.

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 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 for the IDP on LERF contributors.

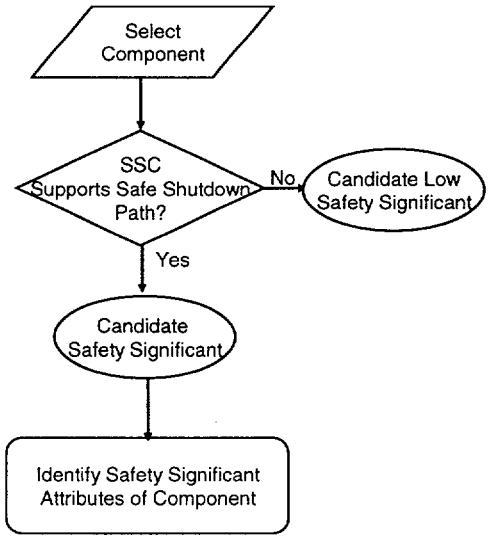
The recommended sensitivity studies for seismic PRA are identified in Table 5-4:

Table 5-4
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 • 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 adequacy

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 5-5.

Figure 5-5
SAFETY SIGNIFICANCE PROCESS FOR
SYSTEMS AND COMPONENTS ADDRESSED IN SEISMIC MARGINS



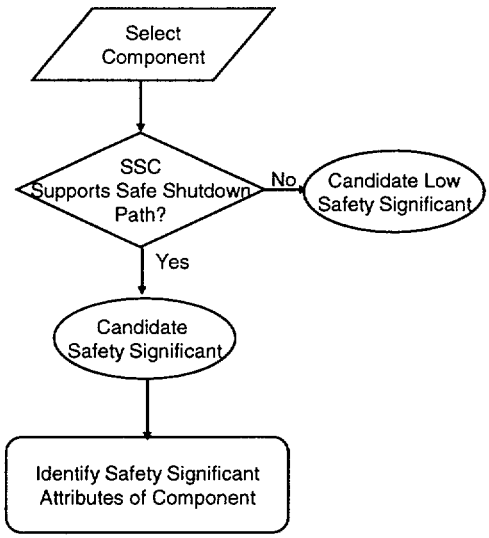
In this process, after identifying the design basis and severe accident functions of the component, the seismic margins analysis is reviewed to determine if the component is credited as part of the safe shutdown paths evaluated. If a component is credited, it is considered safety significant. This is 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.

5.4 Assessment of Other External Hazards

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 Figure 5-3 and discussed below. Plants that relied upon an external hazard screening to assess external hazards for the IPEEE would use the modified process shown in Figure 5-6.

Figure 5-6
OTHER EXTERNAL HAZARDS



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 cannot 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 5-5.

Table 5-5
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 • Set all maintenance unavailability terms to 0.0 • Any applicable sensitivity studies identified in the characterization of PRA adequacy

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 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 recommendations for the IDP on LERF contributors.

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 5-6.

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.

5.5 Shutdown Safety 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 Figure 5-2. Plants that do not have a shutdown PRA would use the modified process shown in Figure 5-7 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.

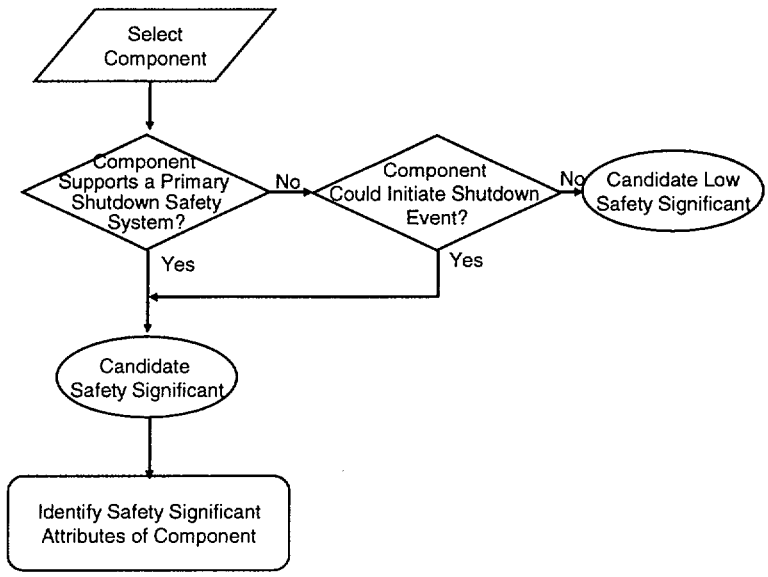
The same sensitivity studies identified in Table 5-2 should be used in the evaluation of shutdown risk significance.

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 5-7.

Figure 5-7

SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS CREDITED IN NUMARC 91-06 PROGRAM



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

- It could initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.),
- 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 as defined by the plant outage risk management guidelines cannot be met for the safety function without the system, structure, or component. The Outage Risk Management Guidelines categorize the level of safety and specify the minimum acceptable number of systems for each safety function.

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 that has the following attributes:

- It has a 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.

5.5 Integral Assessment

In order to provide 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

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

Where,

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

Integrated Risk Achievement Worth Importance

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

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

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 should 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.

6 DEFENSE-IN-DEPTH ASSESSMENT

In cases where the component is 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.

6.1 Core Damage Defense-in-Depth

The initial assessment should consider both the level of defense in depth in preventing core damage and to the frequency of the events being mitigated. Figure 6-1 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. The defense-in-depth matrix is similar in form to the Significance Determination Process used in the Reactor Oversight Process and uses the same concepts of diverse and redundant trains and systems in evaluating the level of defense-in-depth.

For example, if a PWR found that SSCs in the condensate system could be categorized 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 categorization 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 categorization.

6.2 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 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?

Long-term Containment Integrity

- Does the SSC perform a function that is not considered in CDF and LERF, but could be beneficial in preserving long-term containment integrity (i.e., containment temperature or pressure control)?

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.

Figure 6-1

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				

7 PRELIMINARY ENGINEERING CATEGORIZATION OF FUNCTIONS

7.1 Engineering Categorization

This step involves the assignment of a preliminary safety significance to each of the functions identified previously. The safety significant SSCs from the component safety significance assessment (Section 5) are mapped to the appropriate function for which they had a high safety significance. If any SSC function that supports a system function has high safety significance, from either the PRA-based component safety significance assessment (Section 5) or the defense-in-depth assessment (Section 6), then the system function is preliminarily assigned high safety significance. Once a system function has been identified as safety significant, then all components in the flow path (or system segment) supporting that system function are assigned a preliminary safety significant categorization. All other components were assigned a preliminary low safety significant categorization.

Due to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC was assigned the highest risk significance for any function in which that SSC was used.

7.2 Summary of Results

The results of the compilation of risk information and safety significant attributes should be documented for the IDP's use. Figure 7-1 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 system function:

- System name
- The function(s) evaluated.
- The SSCs used as surrogates in the safety significance assessment.
- 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 categorization recommendation to the IDP.

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 7-1

EXAMPLE RISK-INFORMED SSC ASSESSMENT WORKSHEET

System: _____ Function: _____

SSCs Considered in Safety Significance Assessment: _____

		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				

Insights From Sensitivity Studies: _____

Defense-in-Depth Assessment: _____

Recommended Categorization:

Safety Significant: _____ Low Safety Significant: _____

Basis for Categorization: _____

8 RISK SENSITIVITY STUDY

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. 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.

This risk sensitivity study is made using the available PRAs to evaluate the potential impact on CDF and LERF, based on a postulated change in reliability. This 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 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. Risk sensitivity studies should be realistic.

For example, increasing the unreliability of all low safety significant SSCs by a factor of 2 to 5 could 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. Such a degradation is extremely unlikely for an entire group of components. Utility corrective action programs would see a substantial rise in failure events and corrective actions would be taken long before the entire population experienced such a degradation. Individual components may see variations in performance on this order, but it is exceedingly unlikely that the performance of a large group of components would all shift in an unfavorable manner at the same time.

The risk sensitivity study should be performed by manipulating the unavailability terms for PRA basic events that correspond to component that were identified in the categorization process as having low safety significance because they do not support a safety significant safety function. The basic events for both random and common cause failure events should be increased for failure modes expected to be impacted by the changes in special treatment. A 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 an error factor of 3 and 3.5 for an error factor of 10.

This sensitivity study should be performed for each individual plant system as the categorization of its functions is provided to the IDP. A sensitivity study should be performed for the system, and a cumulative sensitivity for all the SSCs categorized using this process. This should provide the IDP with both the overall assessment of the potential risk implications and the relative contribution of each system.

Reducing the unreliability of safety significant 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-categorize some of the candidate low safety significant SSCs as safety significant SSCs.

The results of an initial 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.

9 IDP REVIEW AND APPROVAL

The 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 functions/SSCs.

9.1 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 risk management), 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. Experience, plant knowledge, and availability to attend the majority, if not all meetings, are important elements 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 judgment 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:

- 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 should be documented. A consensus process should be used for decision-making. Differing opinions should 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:

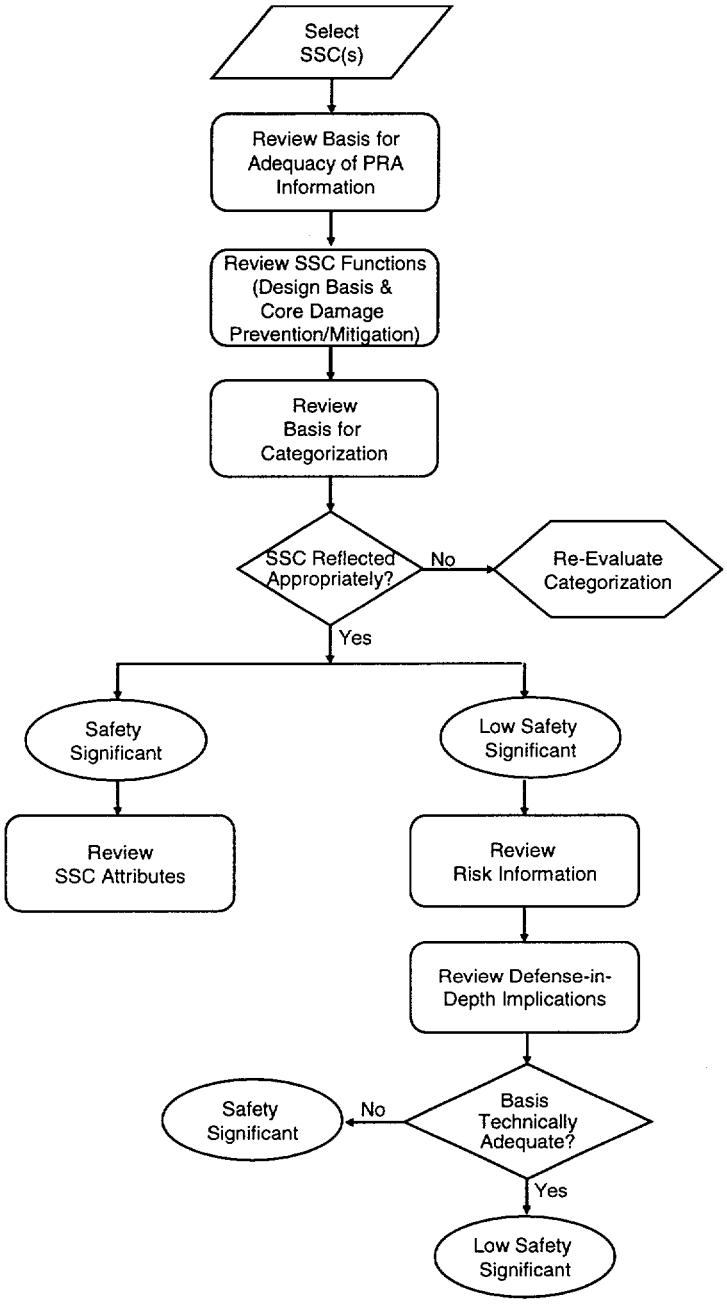
- The designated chairman, panel members, and panel alternates;
- Required training and qualifications for the chairman, members, and alternates;
- Requirements for a quorum, attendance records, agendas, and meeting minutes;
- The decision-making process;
- Documentation and resolution of differing opinions;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

9.2 IDP Process

The preliminary categorization information generated as part of the categorization process, including consideration of the role each function in the plant-specific risk analyses and defense-in-depth, is provided to the IDP for review. The overall functional categorization process to be used by IDP is shown in Figure 9-1.

Figure 9-1

IDP PROCESS



The IDP reviews this preliminary categorization of system functions. In some cases, where the functional role of multiple SSCs is similar, those SSCs may be considered at the same time. For example, the suction and discharge isolation valves on a pump, may have similar functional impacts and could be considered together the pumping function of the system.

The initial steps of the IDP involve review of the primary technical bases for the initial categorization: the basis for adequacy of the PRA results, the system function(s) and the basis for their 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 function has not been appropriately reflected, then it is re-evaluated based on the insights from the IDP.

Review of Safety Significant Functions

For those functions/SSCs determined to be appropriately reflected in the categorization, the IDP should 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 cannot 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 the design basis attributes (for RISC-1), any important to safety attributes (for RISC-2) and any additional attributes that were identified as important to the core damage prevention and mitigation functions of the SSC.

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.

Review of Low Safety-Significant Functions

The IDP's role for these functions is to perform a risk-informed assessment of the SSC categorization including consideration of the risk information, defense-in-depth and safety margins.

Review of Risk Information

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

- Failure of the associated SSC(s) 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 associated SSC(s) 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 function/SSC is necessary for safety significant operator actions credited in the PRA, including instrumentation and other equipment called for in procedures.

- Failure of the function/SSC will result in failure of safety significant functions/SSCs in a manner 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 a function/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 should 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 7);

- 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 7);

- 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:

- Historical data show that these failure modes are unlikely to occur.
- Such failure modes can be detected in a timely fashion.
- Condition monitoring – leading indicators

Functions/SSCs identified as low safety significant in the 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 function/SSC categorized as low safety significance (LSS) to ensure that no significant impacts on safety margin would be expected.

Review of LSS SSCs

The functions/SSCs initially categorized as LSS that include non-safety-related SSCs found in the categorization process to be of low safety significance. The IDP's role for these functions/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) that caused it to be originally classified as important to safety in order for a RISC-4 categorization to be justified. If the IDP concludes that the categorization of the function/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 should be identified to ensure that any core damage prevention and mitigation attributes, that the IDP felt were significant, are included in future treatment.

10 SSC CATEGORIZATION

10.1 Coarse SSC Categorization

When the IDP approves the categorization of system functions, then the initial coarse mapping of components to system function may be used to define the safety significant SSCs. Thus, if a system function is found to be safety significant by the IDP, then all components in the flowpath could be considered safety significant (HSS). In some cases, components may support both safety significant and low safety significant system functions. In these cases, if the SSC is supports for any safety significant system function, then it should be considered safety significant. Likewise, if all system functions supported by the SSC are low safety significant, then the SSC can be considered low safety significant. For some systems, this may be adequate. In other cases, this approach may be found to be too conservative, so a more detailed categorization may be utilized.

10.2 Detailed SSC Categorization

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 importance for the safety-significant SSCs.

There are two options:

1) Assignment of all SSCs in the flow path represented by the function to the RISC classification of that function. While this is a conservative assignment, it may best suit the cost-benefit assessment for Option 2 for a particular system. That is, the effort in going to the next step may not be commensurate with the benefits to be derived.

2) Assignment of SSCs in the flow path represented by the function based on the attributes of the function that the SSC satisfies. This applies primarily to categorizing selected SSCs on high safety significant functions as low safety significant. In this case, the potential failure of an SSC is assessed in light of the safety significant function attributes (e.g., allow flow, prevent flow, prevent fission product releases, etc.). The following criteria can be applied to this process:

   - The criterion for assignment of low safety significance for an SSC in a high safety significant flow path is that its failure would not preclude the fulfillment of the safety significant function. Specific considerations that would permit a low safety significance determination for an SSC in a safety significant functional flow path would include:

- There is no credible active failure mode for the SSC that would prevent a safety significant function from being fulfilled (e.g., a locked open or locked closed valve, a manually controlled valve, etc.),
- An active failure for the SSC would not prevent a safety significant function from being fulfilled (e.g., a vent or drain line not exceeding 1 inch in diameter, SSCs downstream of the first (second?) isolation valve from the active flow path of the function, etc.), and
- Instrumentation that would not prevent a safety significant function from being fulfilled (e.g., radiation monitors that do not have a direct diagnosis function, etc.).

IDP Review of RISC 3 and RISC-4 Components

For SSCs that retain the categorization of the function that they support, only minimal IDP review should be required; there should be no differences from the assessments considered in the initial IDP.

11 CHANGE CONTROL PROCESS

The regulatory change process (10 CFR 50.59) focuses on activities that are directly associated with the 10 CFR 50.2 design bases and that are described in the final safety analysis report.

In a risk-informed regulatory environment, management focus should be on 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 SSC categorization and treatment. Section 6 provides additional details on the change control processes for §50.69. It includes guidance for controlling changes to SSCs and activities that impact beyond design bases function.

11.1 Application of 10 CFR 50.59

10 CFR 50.59 continues to be applied to facility changes as specified in the rule. The first step is to screen the change to determine if the design function is adversely affected.

Change 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 as described in the USFAR. 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 adopts §50.69 should amend its configuration control process to include an additional provision that provides reasonable assurance that the safety-significant beyond design bases function(s) identified in the §50.69 categorization process 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 beyond design bases function(s) identified in the §50.69 process. The information contained in the modification package, the risk-informed categorization process, and the design record file, provide the 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.

For changes that are associated with a safety-significant beyond design basis function(s), the following process is used:

- Perform an engineering evaluation to determine whether there is reasonable assurance that the safety-significant beyond design basis function will be satisfied following the change.
- If a determination is made that the beyond design bases function would be satisfied, the licensee implements the change, and updates the licensee documentation and, as necessary, licensing documentation such as the UFSAR in accordance with NRC requirements.
- If a determination is made that the beyond design basis function would not be satisfied following the change, the licensee has two options:

  (i) Amend the proposed change so that the beyond design basis function would be satisfied, or
  (ii) Evaluate the impact on the §50.69 SSC categorization and the plant specific PRA based on not satisfying the beyond design bases function. Reg. Guide 1.174 provides additional guidance on what may be an acceptable impact on the plant specific PRA and risk to the public.

If the proposed change would result in a change of RISC categorization, the NRC is notified of the change at the same time as a summary of the other §50.59 changes are provided to the NRC.

Design record files and the PRA are updated to reflect the implemented change. 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.

11.2 Changes to Commitments

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

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 commitment is made to update the PRA based on the ASME PRA Standard.

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, Rev 1, Guidelines for Managing NRC Commitment Changes.

Changes in the PRA that result in changes in SSC categorization should be reported to the NRC at intervals consistent with the UFSAR updates.

11.3 Changes To The Plant Specific PRA

The plant specific PRA should be maintained and updated to assure that it reasonably reflects the as-built, as-operated plant is sufficient to support applications for which it being used.

A licensee's configuration control program should monitor changes in the design, operations, maintenance and industry-wide 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 to determine if there is a potential for changing the §50.69 SSC categorization results. (See Reg. Guide 1.174)

11.4 Changes In SSC Categorization

The advancement of technology and the introduction of new information when combined with additional operational experience could impact SSC categorization. This facet is not new. Today, as new information becomes available, licensees may need to adjust safety-related SSC categorizations. Such SSC categorization changes are controlled through the 10 CFR 50.59 process. Similarly, if new information or insights from a PRA update indicate that a SSC is incorrectly categorized, the licensee would take similar actions as it does today.

The extent and scope of any SSC recategorization activities following the implementation of §50.69 may vary dependent upon the specific circumstances, licensing controls and original (safety-related/nonsafety-related) SSC categorization. Recategorization activities should be more demanding for SSCs that are being recategorized from RISC-3/4 SSCs to RISC-1/2 SSCs.

<u>Recategorization of a RISC3 SSC to RISC-1 SSC, or RISC-4 SSC to RISC-2</u>

Advances in technology now enable risk assessments to be performed more efficiently and effectively. These technology improvements provide the industry with a more effective and efficient capability to assess risk and the safety significance of equipment following changes to plant configurations.

Plant modifications, new technical information becomes available, and operating experience increases, introduce the potential for changing the plant specific PRA and the §50.69 SSC categorization results. For the plant specific PRA to be used as a valid assessment tool for regulatory activities, the PRA should be updated at periodic intervals, which could result in changes to SSC categorization.

Changes in CDF, LERF and SSC importance measures provide an indication on whether further evaluations are necessary to determine if there should be a change in SSC

categorization. If further evaluations are necessary, the next step is to determine whether a safety-significant function or a design bases function is affected to the extent that the function would not be satisfied. If there is reasonable assurance that a safety-significant or design bases function can still be satisfied, no immediate action is necessary.

Changes in SSC categorization are not new or limited to plants that have performed §50.69 categorizations. Such changes occur in the deterministic regulatory regime, where licensees change SSC categorization and resolve operability and functional issues in a controlled manner using accepted licensing and work practices and procedures. The same processes that are used in the deterministic regulatory regime should be applied to control and manage changes in the §50.69 SSC categorizations, once an evaluation has confirmed that a RISC-3/4 SSC should be recategorized. The processes involved in these evaluations should include: in-situ dedication, additional engineering analyses and operability determinations. A licensee should follow established licensee procedures if a determination is made that a safety-significant function or design bases function would not be satisfied.

Recategorization of a RISC-1 SSC to RISC-3, or RISC-2 SSC to RISC-4

If new information suggests that a RISC-1 or RISC-2 SSC could be recategorized as a RISC-3/4 SSC, the licensee would follow the same process as described in this guideline for categorizing SSCs. If the §50.69 categorization has been completed for all scheduled SSCs a licensee has the option of using the multi-disciplined station management review committees in place of the IDP to make the final determination on changes in SSC categorization.

12 DOCUMENTATION AND APPROVAL

12.1 Documentation

The documentation on the §50.69 categorization process and the SSCs that have been subject to the categorization process should be stored in a readily retrievable form for use by the licensee and review by the NRC. For SSCs that are included in the new §50.69 categorization scheme by default, i.e., categorized as RISC-1 or RISC-4 SSCs, only a generic reference to the existing SSC categorization needs to be retained.

Documentation relating to the categorization process, including the assumptions and results, should be retained for at least five years after completion of the categorization process, or until the plant specific PRA and, if necessary, the SSC categorization is updated. The documentation should include:

- The plant specific PRA, including the assumptions;
- The comparison and assessment of the plant specific PRA against the PRA quality expectations for this type of application;
- Procedures and guidelines for categorizing the SSCs, including the SSC categorization decision criteria used by licensee staff or contractors in the categorization process;
- References to sources of information and data;
- Integrated Decision-making Panel meeting summaries;
- The results of the SSC categorization and the sensitivity analyses;
- Update of the design record files to documents specific SSC categorization attributes;
- A summary or reference to functional and performance monitoring programs required by §50.69;
- Descriptions and justifications of deviations from this guidance.

These records should be maintained consistent with the licensee's configuration control and documentation management practices. The licensee's design change process should be revised to reflect the availability of new information that should be reviewed as part of change process.

12.2 NRC Review and Approval

A licensee wishing to adopt §50.69 will make a submittal to the Commission requesting approval to implement §50.69 on a specific set of SSCs, as defined by the licensee. The submittal should define the set of regulations that are being adopted. However, it is expected that most licensees will choose to adopt all the regulations in §50.69 (d)(2) that are applicable to RISC-3 SSCs, yet only implement the specific elements on an as needed basis as equipment is changed, maintained and tested. Appendix B provides a submittal outline.

The licensee would notify the NRC of changes in the scope of SSCs or regulations that are being applied to §50.69. Changes in schedule need not be reported to the NRC.

12.3 FSAR Update

A Licensee that adopts §50.69 should update its UFSAR as follows:

- On receipt of NRC approval to proceed with implementing §50.69, the licensee should amend its quality program description included or referenced in the FSAR to include a summary of licensee's industrial program fro low safety-significant SSCs.
- On completion of categorizing the first set of SSCs or system, and on completion of subsequent systems.

These updates should be performed in accordance with NEI 98-03, Guidelines for Updating Final Safety Analysis Reports. The updates would be submitted as part of the regular UFSAR submittal as required by §50.71(e).

13 PERIODIC REVIEW

There are two separate and distinct periodic review elements associated with implementing §50.69: (a) impact from planned SSC categorizations, and (b) periodic reviews following the completion of the §50.69 categorizations.

In case (a), a planned and phased implementation of SSC categorization over several years could result in later SSC categorization activities impacting earlier SSC categorization schemes. As a penultimate step in developing the IDP recommendations on the SSC categorization, a review of the impact of the current categorization activity on previous categorizations should be performed. A determination needs to be made whether the importance measures in the previous categorizations have been changed as a result of these later categorization activities.

The assessment of the impact of later SSC categorizations on the PRA results and earlier categorizations is based on the absolute importance and the new safety-significance determination that are derived from revised SSC importance measure. The absolute importance is the product of the base CDF/LERF and the importance measure (RAW-1/Fussell-Vesely). Categorization reassessments of SSCs that have been previously categorized should be based on the following table:

Table 14-1
IMPACT OF LATER CATEGORIZATION ACTIVITIES

Existing Categorization	New CDF/LERF	New Significance Based on Importance	New Absolute Importance	New Categorization
Low	Higher	Safety-Significant	Higher	Safety-Significant
Low	Reduced/Same	Safety-Significant	Higher	Safety-Significant
Safety-Significant	Reduced/Same	Low	Lower	Low
Safety-Significant	Higher	Low	Lower	Low

In case (b), the periodic review of changes that could impact the SSC categorization following the completion of the 10 CFR 50.69 categorization activities, an evaluation is performed on the SSC categorization impact from changes in equipment performance or the introduction of new technical information. Plant changes that would impact the categorization of SSCs should be prioritized to ensure that the most significant changes are incorporated as soon as practical.

The first step is to determine whether an immediate evaluation is necessary based on the new information. An immediate evaluation and review should be performed if the new information is associated with a RISC-3 or RISC-4 SSC and would have prevented, or did prevent a safety-significant function from being satisfied. If the new information or deficiency would not have inhibited a safety-significant function, then the evaluation

should be performed in a time frame that permits input into the licensee's general PRA update activities.

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[2].

Additional guidance on PRA updates is provided in Section 5 of the ASME PRA Standard.

[2] 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 IDP

14 REFERENCES

1. 10 CFR50.69, *Scope of Structures, Systems and Components, Governed by Special Treatment Requirements*
2. EPRI TR-105396, *PSA Applications Guide*,
3. Regulatory Guide 1.174, *An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*,
4. NRC SECY 99-256, *Rulemaking Plan For Risk-Informing Special Treatment Requirements*,
5. NUMARC 93-01, Rev. 2 *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*
6. NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*
7. NRC Regulatory Guides 1.175, 1.176, 1.177 and 1.178,
8. NRC Reg Guide on PRA Adequacy – Under development
9. Nuclear Energy Institute, "NEI 00-02, Revision 3, *Probabilistic Risk Assessment (PRA) Peer Review Process Guidance*,".
10. NEI 96-07, *Guidelines for 10 CFR 50.59 Safety Evaluations*
11. NEI 97-04, Revision 1, *Design Bases Program Guidelines*
12. NEI 98-03, *Guidelines for Updating Final Safety Analysis Reports*
13. NEI 99-04, *Rev. 1, Guidelines for Managing NRC Commitment Changes*
14. *NEI 00-02, Probabilistic Risk Assessment Peer Review Process Guideline*
15. *NRC letter to NEI dated April 2, 2002, NRC Staff Review Guidance for PRA Results used to support Option 2 Based on NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline, " supported by NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guideline."*
16. ASME Code Case, N658, *Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities*
17. ASME RA-s-2002, *Standard for Probabilistic Risk Assessments for Nuclear Power Plant Applications*

APPENDIX A

GLOSSARY OF SELECTED TERMS

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) - See ASME PRA Standard

Core damage - See ASME PRA Standard

Core damage frequency (CDF) - See 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 - See 10 CFR 50.2

Design functions – See NEI 96-07

Design bases functions - See NEI 97-04

Dependency - See 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 - See ASME PRA Standard

Large early release - See ASME PRA Standard

Large early release frequency (LERF) - See ASME PRA Standard

Probabilistic risk assessment (PRA) - See 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 - See NUMARC 93-01, Rev 2

Risk achievement worth (RAW) importance measure - See ASME PRA Standard

Safety-related structures, systems and components - See 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 - See NUMARC 93-01, Rev 2

APPENDIX B

SUBMITTAL OUTLINE/EXAMPLE

OPTION 2
PROGRAM SUBMITTAL

Owner/Licensee Name

Subject Plant
Unit

NRC Docket Number

NOTE: *Items shown in italics reflect plant-specific information that needs to be provided in an actual Option 2 submittal.*

Option 2 Implementation Plan
Subject Plant
Unit

Table of Contents

INTRODUCTION

The objective of this submittal is to request adjustment to the scope of equipment subject to NRC special regulatory treatment (controls) per the regulatory process prescribed in 10 CFR 50.69, "Scope of Structures, Systems and Components, Governed by Special Treatment Requirements." The assessment and safety categorization of the structures, systems and components referenced in this submittal will be performed in accordance with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" and with Reg. Guide 1.XXX. *Licensee's name and unit number,* takes exception to NEI 00-04 and Reg. Guide1.XXX in the following areas:

- *Licensee lists the exceptions*

The technical basis for these exceptions and the basis for the alternative approach are provided in the Appendix to this submittal.

Background

The intent of the 10 CFR 50.69 regulatory initiative is to adjust the scope of equipment subject to special regulatory treatment (controls) to better focus licensee and NRC attention and resources on equipment that has safety significance. NEI 00-04 uses an integrated decision making process to define the scope of equipment that should be subject to NRC special treatment provisions.

The process identifies and categorizes the set of equipment that is safety-significant by blending risk insights, new technical information and operational feedback. A central task in the implementation of the §50.69 initiative is the use of groups of experienced licensee-designated professionals to make equipment categorization determinations. Treatment is then applied As prescribed in §50.69 consistent with the revised equipment safety categorizations.

SSC SCOPE & APPROACH

Scope of SSCs selected for §50.69 safety categorization assessment

The following systems are the scope of applicability for the implementation of §50.69 at *subject plant, unit,* under this submittal.

- *List the selected systems that are the subject of this approval request and that are being subject to the revised categorization process*

Schedule for Implementation

Provide schedule for implementing SSC categorization

The Director of NRR will be informed of changes to the SSC scope of applicability for §50.69 prior to implementing §50.69 on these systems, or in major changes in the schedule for implementation that result in an extension to the categorization activities, for the systems referenced above, in excess of 12 months.

Approach

The SSCs from the above systems will be placed in four categories as defined by 10 CFR 50.69 using the NRC endorsed NEI 00-04, except as described in the Appendix.

The categorization process uses an integrated decision-making process to determine SSC categorization by blending plant specific risk insights; operational feedback and experience (industrywide and plant specific); and new technical information.

Sensitivity studies will be performed in accordance with NEI 00-04, and the results assessed against the criteria defined in Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.* The impact of changes to SSC categorization and controls will be monitored through periodic PRA updates, as determined by industry consensus standards.

Consistent with Reg. Guide 1.XXX, this submittal, as a risk-informed application, meets the intent and principles of Regulatory Guide 1.174 as described below:

- The Proposed Change Meets the Regulations – The changes in special treatment are made under 10CFR50.69.

- The Proposed Change Is Consistent With The Defense-In-Depth Philosophy – The recategorization and treatment process provides reasonable assurance that safety functions are maintained. Therefore, defense-in-depth will not be impacted. As part of the categorization process, a review is performed which assesses the role the SSC plays in ensuring defense-in-depth.

- The Proposed Change Maintains Sufficient Safety Margins – The recategorization and treatment process provides reasonable assurance that safety-significant functions are maintained. In addition, there will be reasonable confidence that the design bases will be maintained. Therefore, safety margins will not be impacted.

- Any Increases in Core Damage Frequency or Risk Should Be Small and Consistent With the Intent of the Commission's Safety Goal Policy Statement – They are-categorization 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.

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

Integrated Decision-Making Panel (IDP)

A licensee-designated integrated decision-making panel will make the determination on SSC categorization. The IDP will be responsible for oversight of the categorization process, review and approval of SSC categorization, and procedure and working practice development.

Procedures will be developed and approved in accordance with *plant name* procedures to control and document IDP activities and assure consistency in the decision-making process. The IDP panel members are:

- *List panel members, titles, and brief summary of plant/experience*
- *List of procedures*

Application of NRC Special Treatment Requirements

The revised SSC scope will be applied to the following special treatment requirements

- *List the selected NRC special treatment requirements or just reference §50.69.*

Change Control Provisions

The existing regulatory change control provisions prescribed in 10 CFR 50.59, "Changes, Tests and Experiments;" 10 CFR 50.54, "License Conditions;" 10 CFR 50.69; and as amplified in NEI 00-04 will be used to control changes to plant configuration, SSC categorization, and treatment requirements. These measures include a change control process for changes that could impact a beyond design basis function, as described in NEI 00-04. Changes to the PRA will be controlled through the application of NEI 99-04, Revision 1, "Guidelines for Managing NRC Commitment Changes."

CATEGORIZATION BASIS

The Subject Plant has performed a PRA that estimates core damage frequency and large early release frequency due to internally initiated events and internal flooding. Other important risk contributors, such as seismic risk, fire risk, other external event risks (high winds, tornadoes, etc.) during power operation, and risk during outage conditions have also been analyzed using methods that involve use of a PRA to quantify these risk impacts, or may involve simplified analyses or qualitative methods, or a combination of

these methods.

The Subject Plant PRA is capable of quantifying core damage frequency (CDF) and large early release frequency (LERF) for power operations due to internal events and reflects the as-built and as-operated plant.

Plant-Specific Risk Information

The existing CDF and LERF values at the time of preparing this submittal are:

CDF – *Plant specific information*
LERF *Plant specific information*

Other plant specific PRA information should be described, such as:

- *The specific risk analyses to be utilized;*
- *The bases for determining that the analyses are both applicable and useful in categorization*

Characterization of PRA Quality

PRA input into the categorization process includes internal events PRA analyses and risk assessments encompassing external and shutdown events. The *Subject Plant's* PRA meets accepted attributes and characteristics as defined in Reg. Guide 1.XXX and has been subject to the Industry Peer Review Process for PRAs as described in NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance".

The Subject Plant to provide the following information on the 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 of the internal events PRA, including elements that received grades lower than 3.*
- *The disposition of any peer review fact and observations (F&Os) classified as A or B importance.*
- *Provision of information identified in the NRC review of NEI 00-02, NRC letter to NEI dated April 2, 2002,* NRC Staff Review Guidance for PRA Results used to Support Option 2 Based upon NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Supported by NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guideline."

The Subject Plant *provides the following additional information on other PRA Analyses, [If applicable]*

- *A basis for why the other licensee specific PRA analyses (e.g., external events and shutdown) 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.*

DOCUMENTATION UPDATE

The documentation on the § 50.69 categorization process and the list of SSCs that have been subject to the categorization process will be stored in a readily retrievable form for use by the *Subject Plant* and review by the NRC.

Documentation relating to the categorization process, including the assumptions and results, will be retained for at least five years after completion of the categorization process, or until the plant specific PRA and, if necessary, the SSC categorization is updated. These records will be maintained consistent with the *Subject Plant's* configuration control and document management procedure(s) XXXX. *The Subject* Plant's design change process will be revised to reflect the availability of new information that will be reviewed as part of change process.

REFERENCES

1. Reg. Guide 1.XXX, "Guidance for Categorizing Structures, Systems and Components under 10 CFR 50.69."
2. 10 CFR 50.69, "Scope of Structures, Systems and Components, Governed by Special Treatment Requirements"
3. ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications
4. ASME Code Case N-658, *Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities*
5. NRC Regulatory Guide X.XXX PRA Technical Adequacy
6. Regulatory Guide 1.174, *An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*,
7. NRC SECY 99-256, Rulemaking Plan For Risk-Informing Special Treatment Requirements,
8. NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline."

9. NEI 99-04, Revision 1, "Guidelines for Managing NRC Commitment Changes."
10. Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis."
11. NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance."
12. *NRC letter to NEI dated April 2, 2002,* NRC Staff Review Guidance for PRA Results used to Support Option 2 Based upon NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Supported by NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guideline."
13. EPRI TR-105396, PSA Applications Guide,
14. NUMARC 93-01, Rev. 2 Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
15. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management

Appendix to *Licensee's Name and Plant* 10 CFR 50.69 Submittal

Basis and Alternative SSC Categorization Methodology for Exceptions to NEI 00-04 Categorization Process for 10 CFR 50.69

Enclosure 2

Summary of Changes Made to NEI 00-04, Rev B

1. Removed treatment sections
2. Updated Introduction and Background
3. Incorporated system function categorization process and added flowchart developed based on pilots.
4. Updated the change control process.
5. Modified description of PRA quality to address ASME PRA Standard and NRC Option 2 PRA Review Guidance
6. Changed guideline so that SSCs that do not have a role in CDF/LERF are considered low safety significant.
7. Added qualitative treatment of late containment failure as an input to IDP.
8. Added discussion of approach to treating changes in PRA model that potentially changes categorization
9. Added discussion on treatment of implicitly modeled SSCs. This is a major benefit of the revised approach that relies upon system functions as the initial basis for categorization.
10. Modified Figure 2.4-4 to clarify safety significance categorization and address NRC comment
11. Deleted references to monitoring as part of categorization
12. Added discussion of treatment of fire barriers and fire suppression systems to address NRC comment
13. Added guidance to document cases where IDP reviewed preliminary categorization and decided to re-categorize the SSC