NUCLEAR ENERGY INSTITUTE

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November 6, 2003

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 D to NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline.* The document has been revised to address the issues identified by NRC's August 14, 2002 transmittal of draft Regulatory Guide DG-1121. Other revisions have been implemented to reflect lessons learned from pilot applications of the guidance, and publication of the proposed rulemaking. Both a line-in, line-out version, reflecting changes from Revision C, and a final version are enclosed.

The significant changes include:

- A new Section 1.5, "Categorization Process Summary" that discusses how the categorization process addresses risk due to internal and externally initiated events, and risk in plant shutdown conditions.

- Revisions to Section 3.3, "Characterization of the Adequacy of Risk Information" to address NRC draft regulatory guide DG-1122 regarding demonstration of PRA technical adequacy.

- Revisions to Section 5.1, "Internal Event Assessment" to address common cause risk achievement worth (RAW) consideration, based on discussions with NRC staff in public meetings.

- Revisions to Section 6.1, "Core Damage Defense in Depth" to provide more information on the purpose and use of Figure 6-1 to address defense in depth for RISC-3 SSCs.

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Add: David Matthew

- Combination of Sections 11 and 12, now "Program Documentation and Change Control" to better align with the structure of the proposed rule.

- Additional information relative to conduct of Periodic Reviews (now Section 12).

We believe this revision addresses all issues that have been identified by NRC in reviews of previous drafts. Industry pilot efforts are proceeding with the use of this draft guideline to support the categorization process and conduct of the integrated decision panel at one pilot plant next month.

Our intent is for the final version of DG-1121 to achieve NRC endorsement of the guidance without exception. We would be happy to meet with the staff to discuss the changes to the guidance. If NRC staff has any questions, please contact Biff Bradley (202)-739-8083, e-mail reb@nei.org, or me.

Sincerely

Anthony R. Pietrangelo

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Enclosures

Cc: Ms. Suzanne C. Black
Mr. Michael R. Johnson

NEI 00-04 (DRAFT - Revision D)

10 CFR 50.69
SSC Categorization
Guideline

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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

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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 with the 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.

The categorization process described in this section is one acceptable way to undertake the categorization of SSCs. Other methods using a different combination of probabilistic and deterministic approaches and criteria can be envisioned. However, it is expected that the guiding principles (Section 1.3) of this guidance would be maintained. Licensees wishing to use a different method for categorizing SSCs using risk-informed insights need to submit the methodology for NRC review and approval.

[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.

Changes to this guideline are controlled through the normal regulatory change control processes. Section 11 provides guidance on program documentation and change control.

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 could occur, 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

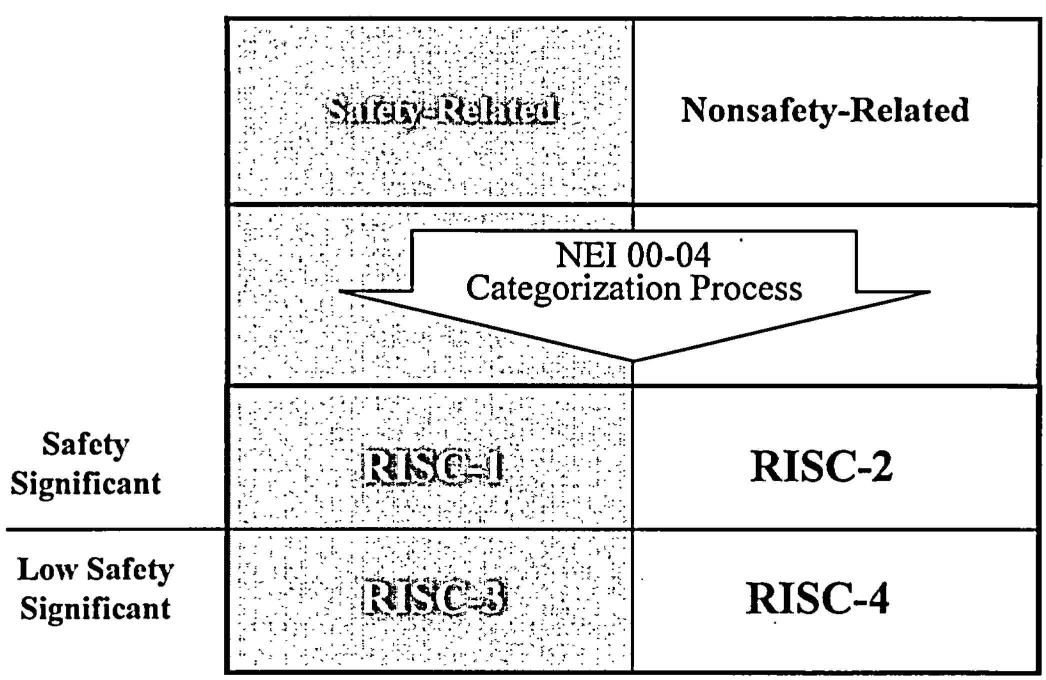
resources on equipment that has safety significance. This guideline addresses the use of risk insights to define the scope of equipment that should be subject to NRC special treatment provisions as defined in §50.69.

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 augmented 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 safety related SSC will be re-categorized as RISC-1 unless a basis can be developed for re-categorizing it as RISC-3.
- 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 an outstanding safety performance record. The existing NRC regulations together with the NRC's regulatory oversight and inspection processes clearly provide adequate protection of public health and safety. As a result, the decision to 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.

1.5 CATEGORIZATION PROCESS SUMMARY

The NEI 00-04 categorization process embodies the principles of risk-informed regulation described in Reg. Guide 1.174 (Figure 1-2). The plant-specific risk analyses provide an initial input to the process. SSCs identified as high safety significant (HSS) by the risk characterization process are identified for an integrated decision-making panel (IDP). The IDP cannot re-categorize an SSC identified by the risk analysis as HSS. The IDP function is to review the assessment and assure that the system functions and operating experience have been appropriately considered in the risk analyses.

SSCs that are safety related and considered to be low safety significant (LSS) based on the plant-specific risk analyses are evaluated in a defense-in-depth characterization process. This deterministic process addresses the role of the SSC with respect to both core damage prevention and containment performance. If defense-in-depth characterization identifies that the SSC should be considered HSS, then it is re-categorized as HSS and recommended to the IDP as a RISC-1 SSC. Here again, the IDP cannot re-categorize an SSC identified by the as HSS. The IDP function is to review the assessment and assure that the system functions and operating experience have been appropriately considered.

If an SSC is found to be LSS by both the risk categorization process and the defense-in-depth characterization process, then it is recommended to the IDP to be LSS. The IDP reviews the categorization process applied to the SSC and, if the IDP feels that the operating experience or functions merit a HSS categorization, they can re-categorize it.

Thus, only if an SSC is found to be of low safety significance by all three (i.e, the risk characterization process, the defense-in-depth characterization process and IDP review), will it be categorized as low safety significant.

Risk Characterization

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

- Internal Event Risks
- Fire Risks
- Seismic Risks
- Other External Risks (e.g., tornados, external floods, etc.)

- Shutdown Risks

Separate evaluation is appropriate to avoid reliance on a combined result that fails to address these differences.

Table 1-1 provides a summary of the alternative approaches taken to address each risk contributor. A brief description of each of these aspects is described.

Internal Event Risks

A high quality PRA is required for the categorization of SSCs relative to internal events, at-power risks. Importance measures related to Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) are used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2). In addition, several sensitivity studies are defined which exercise key areas of uncertainty in the PRA (e.g., human reliability, common cause failures, and no maintenance plant configuration). If an SSC that had been initially identified as low safety significant is found to exceed the safety significance thresholds in a sensitivity study, this information is provided to the IDP, along with an explanation of why the sensitivity study identified the SSC to be safety significant.

Fire Risks

A fire risk analysis, either a plant-specific fire PRA or a Fire Induced Vulnerability Evaluation (FIVE) analysis that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to fire risks. If a fire PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the fire risk contribution is shown to be sufficiently small (in comparison to the internal events risk) as to make the overall safety significance of the SSC low (RISC-3 or -4) in the Integrated Importance Assessment (see below). Sensitivity studies, including fire-specific sensitivity studies, are also identified and used in a similar manner.

In the event a FIVE analysis is used, the categorization process is necessarily more conservative (i.e., designed to identify more SSCs as safety significant). This is due to the fact that FIVE is a screening tool. As such, the resulting scenarios and frequencies have an uneven level of realism. Thus, importance measures are not an effective means for identifying safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the mitigation of any unscreened fire scenario (i.e., retained for consideration in the FIVE analysis) as safety significant. In addition, all screened scenarios are reviewed to identify any system functions and associated SSCs that would result in a scenario being unscreened, if that system function was not credited. This measure of safety significance assures that the SSCs that were required to maintain low fire risk are retained as safety significant.

Figure 1-2
Summary of NEI 00-04 Categorization Process

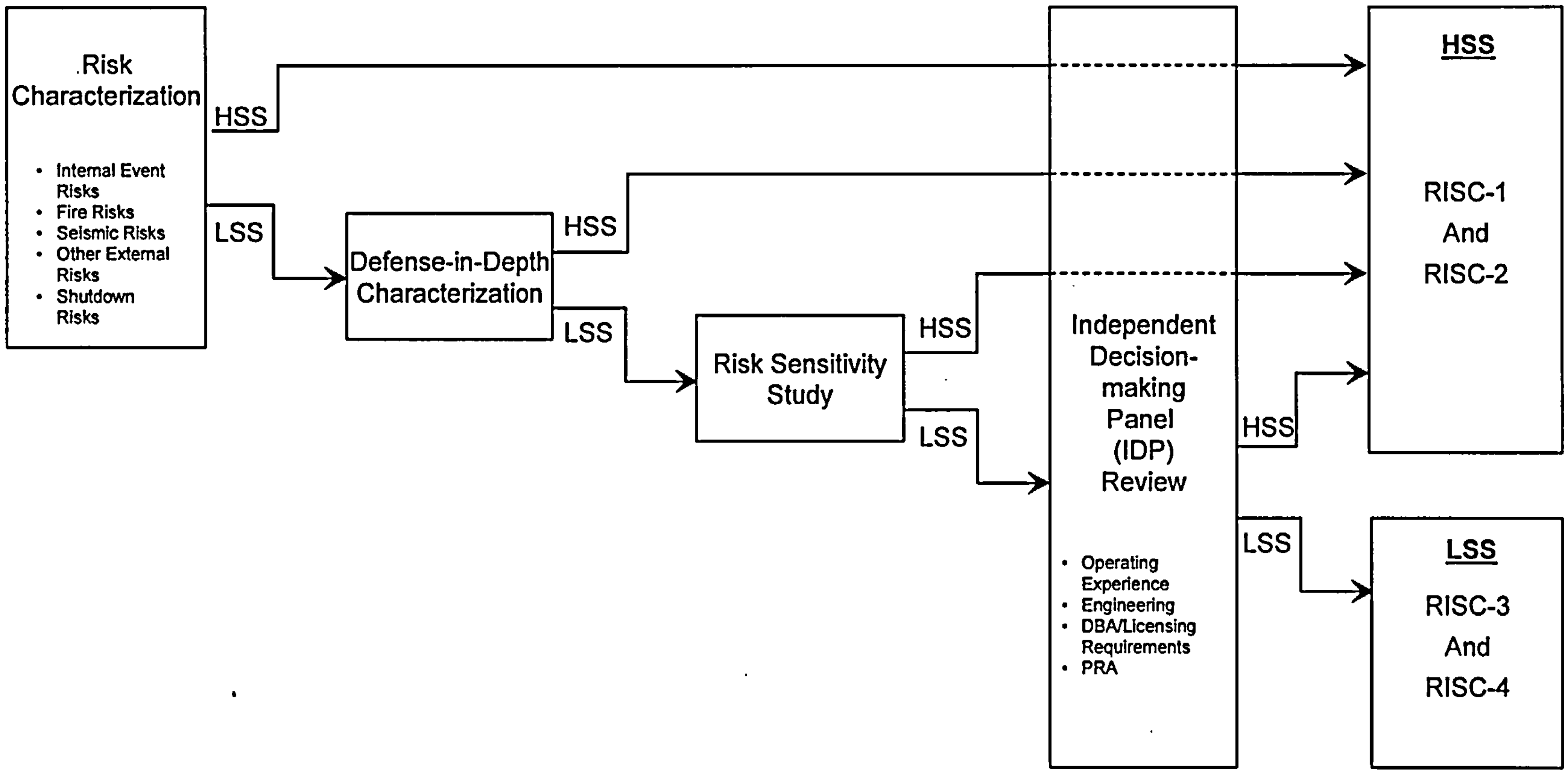


Table 1-1
Summary of Risk Significance Characterization Used in NEI 00-04

Risk Source	Alternative Approaches	Scope of Safety Significant SSCs
Internal Events	PRA Required	Per PRA Risk Ranking
	Screening Approaches Not Allowed	n/a
Fire	Fire PRA	Per PRA Risk Ranking
	FIVE (Fire Induced Vulnerability Evaluation)	All SSCs Necessary to Maintain Low Risk
Seismic	Seismic PRA	Per PRA Risk Ranking
	SMA (Seismic Margins Analysis)	All SSCs Necessary to Maintain Low Risk
High Winds, External Floods, etc.	PRA	Per PRA Risk Ranking
	IPEEE Screening	All SSCs Necessary to Protect Against Hazard
Shutdown	Shutdown PRA	Per PRA Risk Ranking
	Shutdown Safety Plan	All SSCs Required to Support Shutdown Safety Plan

Seismic Risks

A seismic risk analysis, either a plant-specific seismic PRA or a seismic margin analysis (SMA) that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to seismic risks. If a seismic PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the seismic risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (RISC-3 or -4) using the integrated importance assessment. Sensitivity studies, including seismic-specific sensitivity studies, are also identified and used in a similar manner.

In the event an SMA is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant). This is due to the fact that SMA is a screening tool. As a screening tool, importance measures are not available to identify safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the seismic margin success paths as safety significant. This measure of safety significance assures that the SSCs that were required to maintain low seismic risk are retained as safety significant. The seismic PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but the SMA identifies them as safety significant regardless of their capacity, frequency of challenge or level of functional diversity.

Other External Risks

For other external event risks, either a plant-specific external event PRA or a screening analysis that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to other external risks. If an external hazard PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the other external hazard risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (see integrated importance assessment below). Sensitivity studies are also identified and used in a similar manner.

In the event an screening analysis is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant). The NEI 00-04 approach identifies all system/structure functions and associated SSCs that are involved in protecting against the external hazard as safety significant. An example might be a tornado missile barrier. Using a PRA, some barriers might be found to be of low safety significance, depending on the site-specific frequency of tornadoes and the equipment protected by the barrier. Using a screening method, the barrier would be identified as safety significant without regard to those other factors. This measure of safety significance is much more restrictive than the importance measures used in the external hazard PRA and would be expected to yield a larger set of safety significant SSCs than the external hazard PRA. The PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but

the screening approach identifies them as safety significant regardless of their capacity, frequency of challenge or level of functional diversity.

Shutdown Risks

A shutdown risk analysis, either a plant-specific shutdown PRA or a shutdown safety management plan that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to shutdown risks. If a shutdown PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the shutdown risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (see integrated importance assessment below). Sensitivity studies, including shutdown-specific sensitivity studies, are also identified and used in a similar manner.

In the event a shutdown safety management plan is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant) than a plant specific PRA. This is due to the fact that the shutdown safety management plan provides safety function defense in depth without regard to the likelihood of demand or reliability of the functions credited. The NEI 00-04 approach identifies all SSCs necessary to support primary shutdown safety systems as safety significant. This measure of safety significance assures that the SSCs that were required to maintain low shutdown risk are retained as safety significant. The shutdown PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but the shutdown safety management plan approach identifies them as safety significant regardless of the frequency of challenge or level of functional diversity.

Integrated Importance Assessment

Each risk contributor is initially evaluated separately due to the significant differences in the methods, assumptions, conservatisms and uncertainties associated with the risk evaluation of each. In general, the quantification of risks due to external events and non-power operations tend to contain more conservatisms than internal events, at-power risks. As a result, performing the categorization simply on the basis of a mathematically combined total CDF/LERF would lead to inappropriate conclusions. However, it is desirable in a risk-informed process to understand safety significance from an overall perspective, especially for SSCs that were found to be safety significant due to one or more of these risk contributors.

In order to facilitate an overall assessment of the risk significance of SSCs, an integrated computation is performed using the available importance measures. This integrated importance measure essentially creates a weighted-average importance based on the importance measures and the risk contributed by each hazard (e.g., internal events, fire, seismic PRAs). The weighted importance measures can be significantly influenced by the relative contribution of the hazard. For example, an SSC that is very important for a

hazard that contributes only 1% to the total CDF/LERF would be found to have very low importance measures when the integrated assessment is performed. In no case will the integrated importance measure be larger than the largest of the individual hazard importance measure. This integrated assessment allows the IDP to determine whether the safety significance of the SSC should be based on the significance for that individual hazard or from the overall integrated result, avoiding a strict reliance on a mathematical formula that ignores the significant dissimilarities in the calculated risk results.

Defense in Depth Characterization

For safety related SSCs initially identified as low safety significant (RISC-3) from the results of the risk significance categorization, an additional defense-in-depth assessment is performed. The defense in depth assessment is based on a set of deterministic criteria based on design basis accident considerations to assure that adequate redundancy and diversity will be retained. This assessment evaluates the SSC functions with respect to core damage mitigation, early containment failure/bypass, and long term containment integrity. If one of these SSC functions is found to be safety significant with respect to defense-in-depth, then it is considered safety significant and re-categorized as safety significant (RISC-1) for presentation to the IDP.

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. This risk sensitivity study is performed using the available PRAs to evaluate the potential impact on CDF and LERF, based on a postulated change in reliability. In this risk sensitivity study, the unreliability of all low safety significant SSCs is increased simultaneously by a common multiplier as an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of all low safety significant SSCs. A simultaneous degradation of all SSCs 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 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. 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, actual net increase in risk.

In cases where the licensee does not use a PRA in the categorization process, the sensitivity study remains a viable indication of potential limiting risk increases. This is due to the fact that the categorization processes for hazards that do not have a PRA is done in a manner that assures the risk sensitive SSCs are categorized as safety significant. For example, in the event a seismic margins analysis (SMA) is used for the categorization, all of the SSCs necessary to maintain the current risk levels are considered

safety significant. As a result, there would not be any change in the treatment for the SSCs that are credited in mitigating seismic risk.

Integrated Decision-making Panel Review

The Integrated Decision-making Panel (IDP) is a multi-discipline panel of experts that reviews the results of the initial categorization and finalizes the categorization of the SSCs/functions. The purpose of the IDP is to assure that the appropriate considerations from plant design and operating practices and experience are reflected in the categorization input.

The IDP considers the safety significance of the SSCs based on:

- the PRA assessments and sensitivity studies,
- a defense in depth assessment from an operational perspective,
- insights from other risk informed programs (e.g., Maintenance Rule, Risk Informed ISI, etc.), and
- operational and maintenance experience.

In order for an SSC/function to be recommended to the IDP as low safety significant, it must have been identified as low safety significant from the perspective of

- Internal Event Risks
- Fire Risks
- Seismic Risks
- Other External Risks
- Shutdown Risks

If it is an SSC that is currently safety related, then the defense in depth assessment must also have shown that the SSC is not safety significant. Finally, the risk sensitivity study verifies that the combined impact of a postulated simultaneous degradation in reliability of all low safety significant SSCs would not result in a significant increase in CDF & LERF.

If an SSC is only identified as safety significant based on a non-internal events PRA (and was not found to be significant in the integrated importance assessment), or by one of the mandatory sensitivity studies, then the IDP will be presented the results and will use other knowledge and experience to decide whether the SSC should be safety significant.

The IDP will not over-rule the categorization process to make an SSC/function low safety significant when the process identifies it as safety significant (i.e., will not move it from RISC-1 to RISC-3). The IDP may, however, identify that the SSC/function was not appropriately reflected in engineering assessment which may result in a new categorization, based on a revised evaluation.

Conclusions

The categorization methodology used to define the low safety significant SSCs, as described in NEI-00-04, assures any reduction in component reliability as a result of changes in treatment will have a negligible impact on plant risk. This degree of assurance is provided by a multi-layered approach to identifying the low safety significant SSCs that includes PRA, deterministic assessments and engineering judgment. In addition, two different plant organizational functions (engineering and the IDP) perform assessments from their own unique perspective. In either the engineering or the IDP assessment, if any of these three elements indicates that an SSC is safety significant, then that categorization (safety significant) is assigned.

In terms of the scope of the PRA used in the risk assessment portion of the categorization process, a reasonable degree of confidence that risk significant SSCs will be appropriately identified can be maintained with a quality internal events at-power PRA. Screening assessments for other initiating events and other modes of operation identify the SSCs necessary to maintain low risk.

The number of independent criteria that an SSC must satisfy in order to be categorized as low safety significant provides a high level of assurance that only SSCs that are truly low safety significant will be categorized as such.

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.

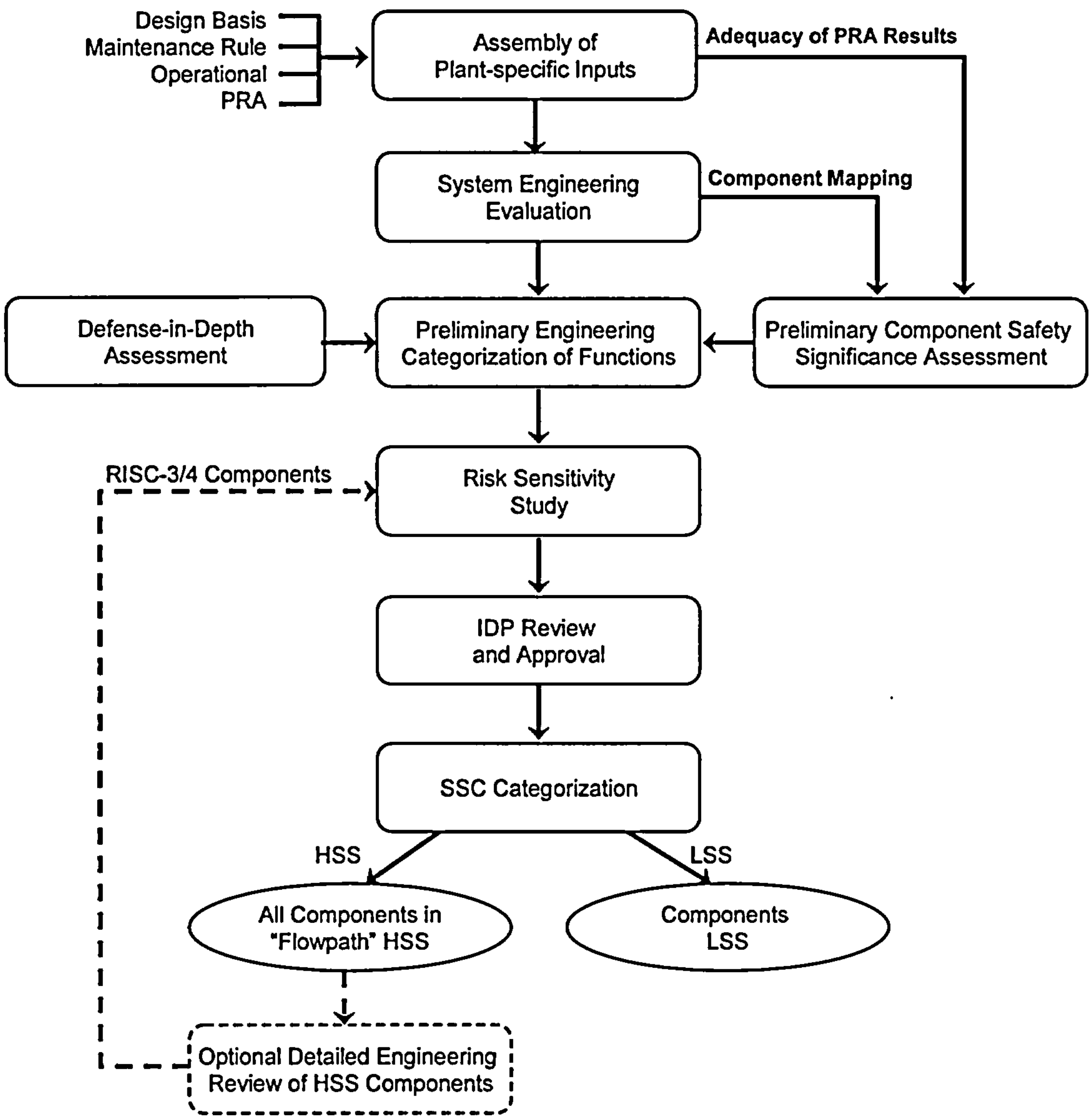
Assembly of Plant-Specific Inputs

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 risk information 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



<u>Component Safety Significance Assessment</u>

This step involves the use of the plant-specific risk information to identify components that are candidate 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 identifies any HSS SSCs that can be categorized as LSS results of that re-catgegorization are 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 is 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 Risk 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 satisfy 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.

When risk information is used to provide insights into the integrated decision-making panel, it is expected that the risk information will have been subject to quality measures. The following describes methods acceptable to ensure that the risk information is of sufficient quality to be used for regulatory decisions and meets the quality standards described in Reg. Guide 1.174:

- 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 the adequacy of the risk information used in the categorization process (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 risk information can be used to support the categorization process, provided it can be shown that the appropriate quality provisions have been met.

Other 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 Risk Information

Figure 3-1 depicts the approach to be employed in demonstrating the adequacy of risk information used in the categorization of SSCs. The adequacy of the risk information builds upon the efforts to review and evaluate the adequacy of the plant-specific internal event full power PRA. There are two options for demonstrating the adequacy of the results of the internal events PRA for use in the categorization process.

The first approach is to utilize the industry peer review process (NEI 00-02). In a letter dated April 24, 2000, NEI requested the NRC staff review the suitability of the peer review process described in NEI 00–02 to address PRA quality issues for this application. NRC issued a request for additional information on September 19, 2000, to which NEI responded by letter dated January 18, 2001. By letter dated April 2, 2002 (ADAMS accession number ML020930632), the NRC staff sent to NEI draft staff review guidance that was developed as a result of its review of NEI 00–02, for intended use for § 50.69 applications.

The staff review guidance is for a focused review of the plant-specific PRA based on a review of NEI 00–02 and NEI 00–04. In order to reach the conclusion that the PRA results support the proposed categorization, the review guidance is structured to lead the staff reviewer to either look for evidence that the impact of a given peer review issue on

PRA results has been adequately addressed in the peer review report and, when necessary, has been identified for consideration by the IDP, or to request further information from the licensee.

If a licensee decides to utilize the NEI 00-02 peer certification to demonstrate the adequacy of the PRA results, the staff review guidance would be used to identify and address potential issues prior to use of the PRA.

The second approach would rely upon the process currently described in draft regulatory guide DG–1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." This guide provides guidance on the NRC position on voluntary consensus standards for PRA (in particular on the ASME standard for internal events PRAs) and industry PRA documents (*e.g.*, NEI 00–02, "Probabilistic Risk Assessment Peer Review Process Guideline"). Ultimately, this guide will be modified to address PRA standards on fire, external events, and low power and shutdown modes, as they become available. The NRC has also developed a draft supporting Standard Review Plan, SRP 19.1, to provide guidance to the staff on how to determine whether a PRA providing results being used in a decision is technically adequate.

If a licensee decides to utilize the DG-1122 process to demonstrate the adequacy of the PRA results, it would be used to identify and address potential issues prior to use of the PRA in support of any 50.69 application.

Both processes rely upon peer review findings as a significant measure of the adequacy of the PRA results. All significant peer review findings will 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 risk information used in the categorization process, such as Fire PRAs, FIVE, Seismic PRAs, SMAs 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 risk information to be used should be documented in a characterization of the adequacy of the PRA. This characterization will be provided to the IDP as a basis for the adequacy of the risk information used in the categorization process and will be summarized in the submittal to the NRC. At a minimum, this characterization should include the following:

Full Power Internal Events PRA

- A basis for why the internal events PRA reflects the as-built, as-operated plant.
- A high level summary of the results of the peer review of the internal events PRA including elements that received grades lower than 3, if NEI 00-02 is used, or lower than ASME Capability Category II, if the DG-1122 process is used.
- The disposition of any significant peer review findings.
- Identification of and basis for any sensitivity analyses necessary to address identified findings.
- Considerations identified by the NRC in their letter to NEI [Ref. 15], if the NEI 00-02 process is used.

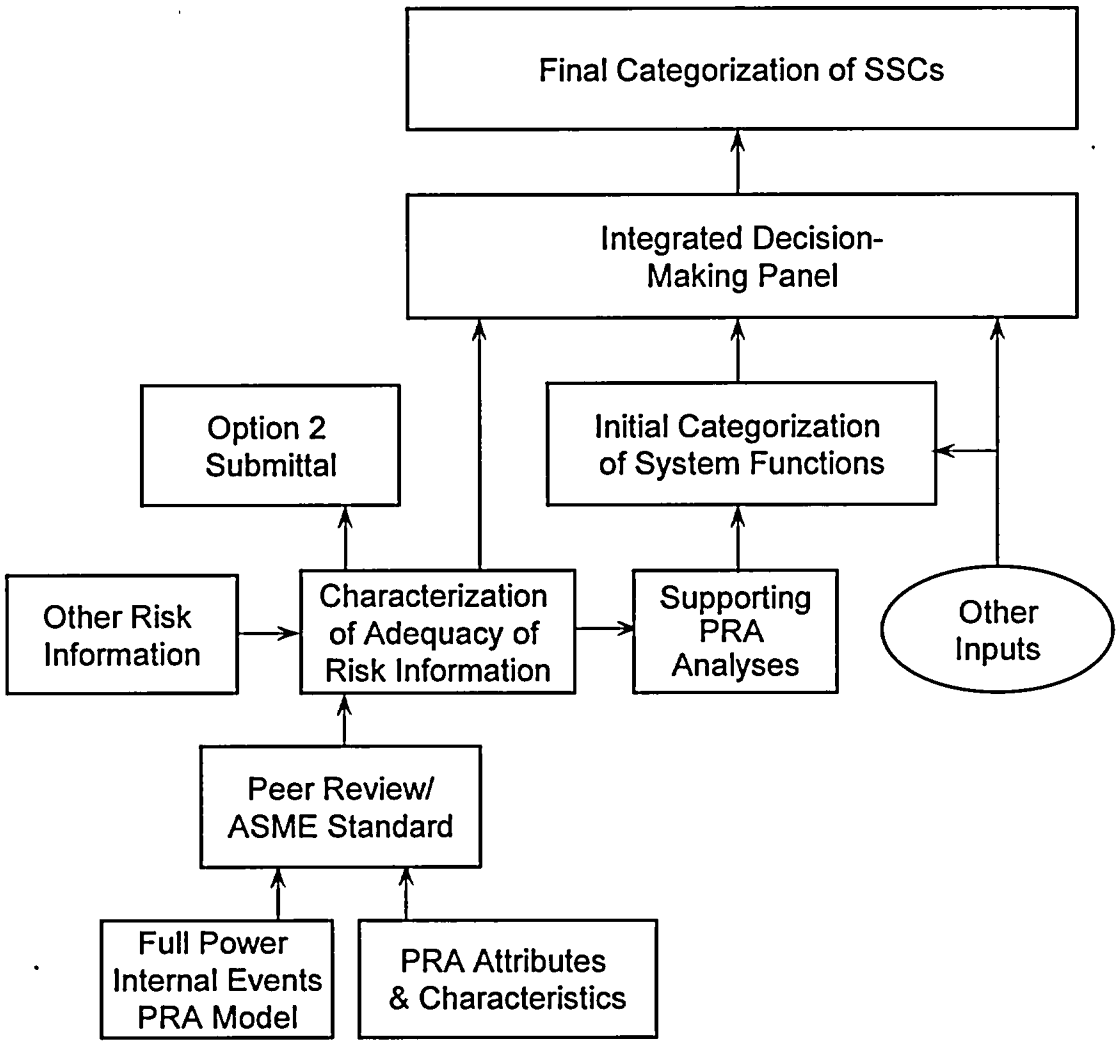
Other Risk Information (including other PRAs and screening methods)

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

The Integrated Decision-making Panel (IDP) should use this information, in combination with the results of the categorization analyses and other information, to finalize the categorization of each function/SSC. The process to be used to justify the adequacy of the risk 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 Considered in the Plant-specific Risk Information
- 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-660, *Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities* (Ref. 16), is being implemented at the same time. [2]

   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.

[2] If this code case is not endorsed at the time of submittal, then the licensee will describe the process to be used in the Option 2 submittal.

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.

Significance 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 candidate 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 event sometimes used in PRAs. The term "evaluated" means:

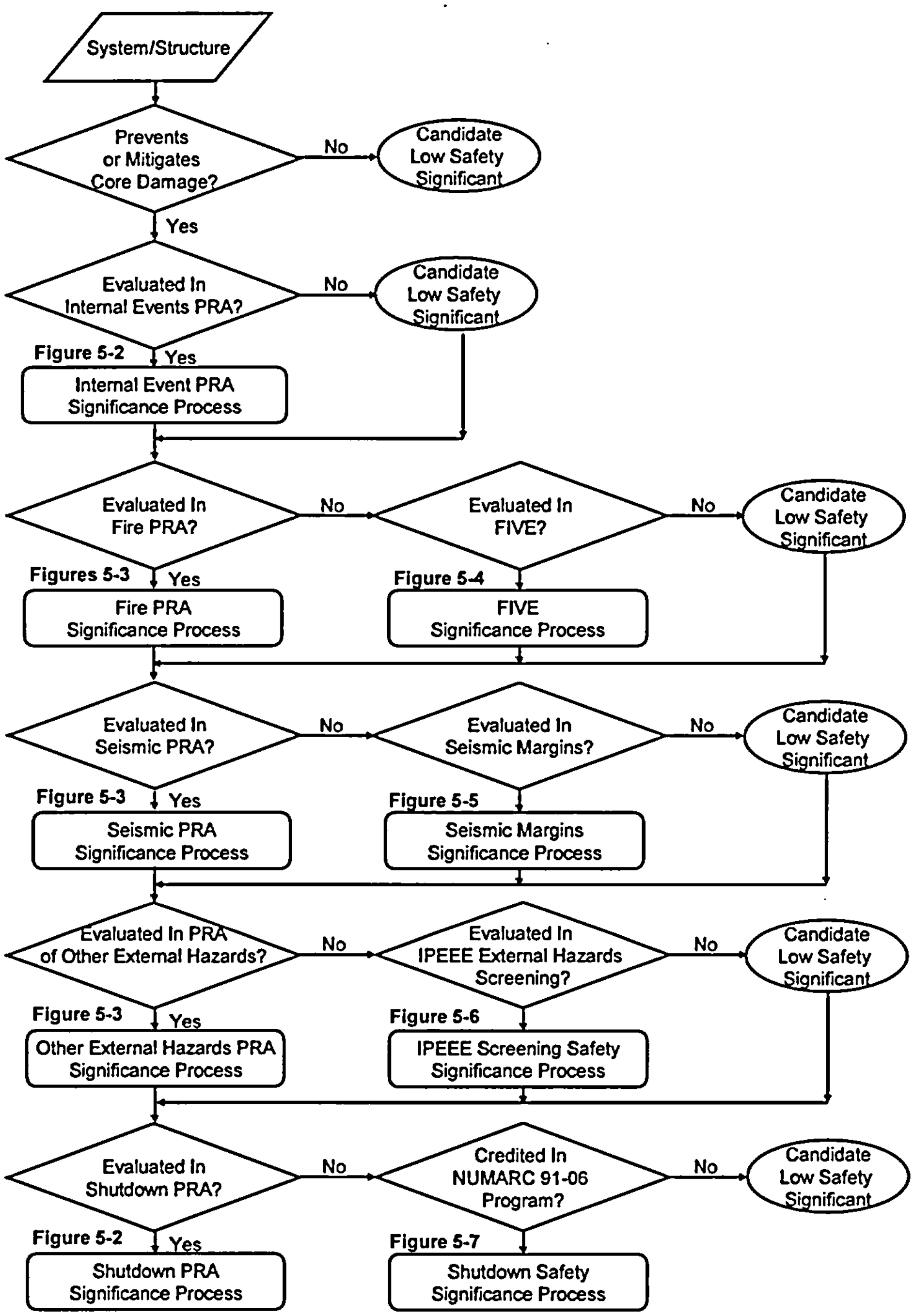
- Can it produce a potential initiating event?
- Is it credited for prevention of core damage or large early release?
- Is it necessary for another system or structure evaluated in the PRA to prevent an event or mitigate an event?

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 SSC is categorized as candidate low safety significant from the standpoint of internal event risks. The evaluation is continued with fire risk.

Figure 5-1

USE OF RISK ANALYSES FOR SSC CATEGORIZATION



Significance 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 they 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 SSC is categorized as candidate low safety significant from the standpoint of fire risks.

Significance 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 SSC is categorized as candidate low safety significant from the standpoint of seismic risk.

Significance 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 SSC is categorized as candidate low safety significant from the standpoint of other external risks.

Significance 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 SSC is categorized as candidate low safety significant from the standpoint of shutdown risk.

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.

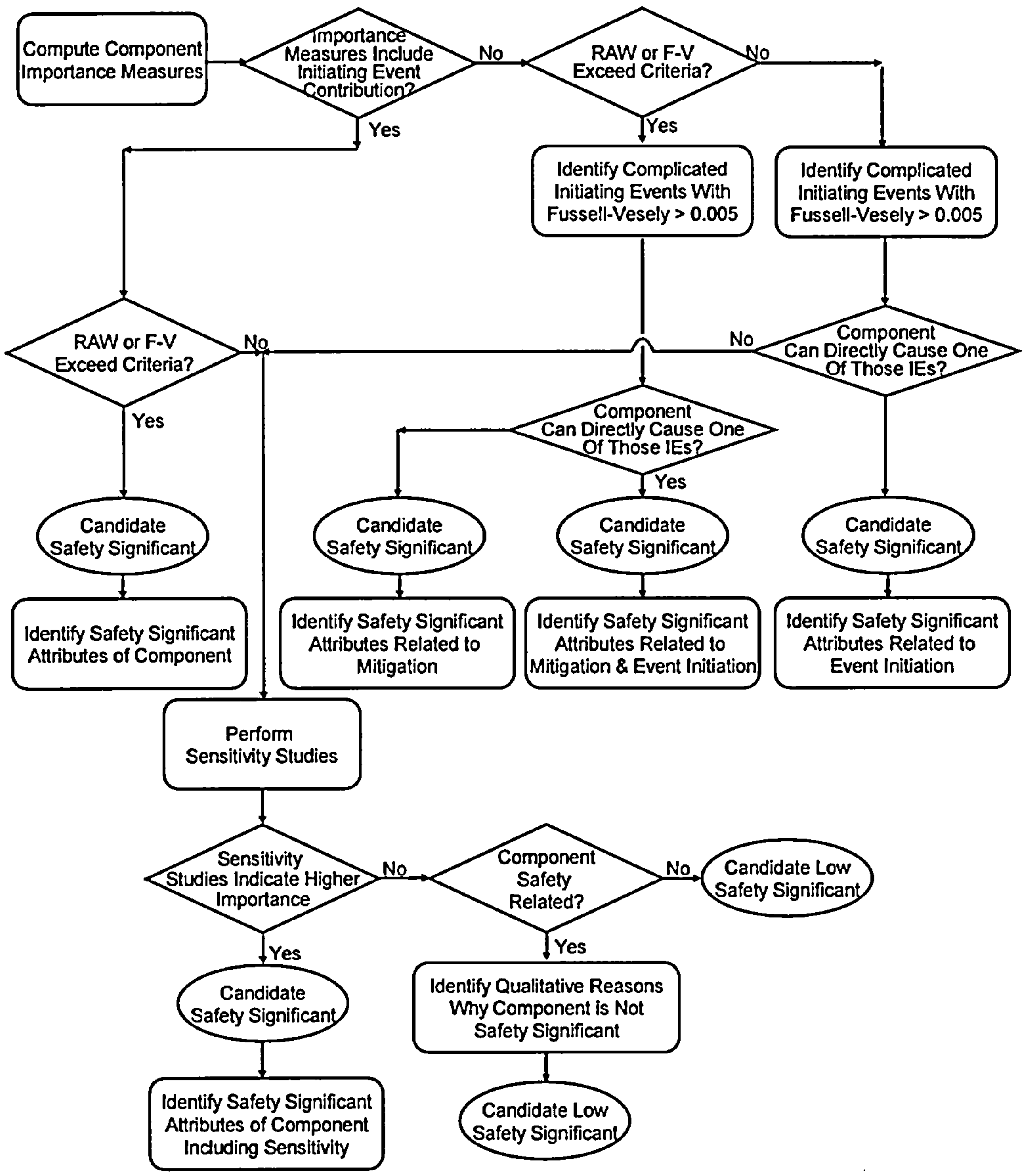
The assessment of importance for an SSC involves the identification of PRA basic events that represent the SSC. This can include events that explicitly model the performance of an SSC (e.g., pump X fails to start), events that implicitly model an SSC (e.g., some human actions, initiating events, etc.) 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. For obtaining risk insights from the PRA for passive pressure boundary components, additional guidance is provided in ASME Code Case N-660, 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[3].

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 because 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 can be expected to be affected by the special treatment requirements being evaluated.

[3] If these code cases and methods are not endorsed at the time of submittal, then the licensee will describe the process to be used in the Option 2 submittal.

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 failure modes of the individual component. In the case of RAW, the common cause event is considered using a different criterion than the individual component RAW. The RAW for common cause events reflects the relative increase in CDF/LERF that would exist if a set of components or an entire system was made unavailable. As a result, the risk significance of the RAW values of common cause basic events are considered separately from the basic events that reflect an individual component. As with the individual component RAW values, if the component being evaluated is included in more than one common cause basic event, the maximum of the common cause RAW values is used to evaluate the significance.

The importance measure criteria used to identify candidate safety significance are:

- Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005
- Maximum of component basic event RAW values > 2
- Maximum of applicable common cause basic events RAW values > 20.

If any of these criteria are exceeded it is considered candidate safety significant.

For example, a motor operated valve may have a number of basic events associated with it (e.g., "failure to open" and "failure to close"), 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). Some SSCs perform multiple functions (e.g., circuit breakers can perform a function necessary for pump operation and a function necessary to protect the bus in case of a fault. In these cases, basic events should be mapped to the appropriate functions so that the significant functions can be identified.

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	CCF RAW
1) Valve 'A' Fails to Open	0.002	1.7	n/a
2) Valve 'A' Fails to Remain Closed	0.00002	1.1	n/a
3) Valve 'A' In Maintenance (Closed)	0.0035	1.7	n/a
4) Common Cause Failure of Valves 'A', 'B' & 'C' to Open	0.004	n/a	54
5) Common Cause Failure of Valves 'A' & 'B' to Open	0.0007	n/a	5.6
6) Common Cause Failure of Valves 'A' & 'C' to Open	0.0006	n/a	4.9
Component Importance	0.01082 (sum)	1.7 (max)	54 (max)
Criteria	> 0.005	>2	>20
Candidate Safety Significant?	Yes	No	Yes

In the above example, Valve 'A' would be considered candidate safety significant on two bases, either one would be sufficient to identify the component as candidate safety significant. The total Fussell-Vesely exceeded the criterion of 0.005 and the RAW criterion was also met for the common cause group including Valve 'A'. Thus, both Valve 'A', Valve 'B' and Valve 'C' would be identified as candidate safety significant due to this criterion. The component failure mode which contributes significantly to the importance of Valve 'A' is failure to open (failure modes 1, 4, 5 and 6). 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.

In cases where the internal events core damage frequency is dominated by an internal flooding result that has a conservative bias, it is appropriate to break the evaluation of importance measures into two steps. This prevents the conservative bias of the flooding analysis from masking the importance of SSCs not involved in flood sceanrios. 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 being 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, including sensitivity studies, 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. In addition, the truncation level used should be sufficient to identify all the SSCs with RAW>2. 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 is reasonable. In linked fault tree PRAs that do not use pre-

solved cutsets, the truncation limit should be evaluated to ensure that converged importance measures are being used. If the model relies on a pre-solved set of cutsets to calculate CDF, then the RAW values may be underestimated and the nominal truncation level may not be capable of identifying all the RAW>2 SSCs, even in a converged solution. Therefore, the truncation of pre-solved set of cutsets should be checked to ensure that the CDF and LERF solutions are sufficiently adequate by justifying the omitted SSCs with RAW>2. In some cases, this may be best handled by complete re-solution of the model without credit for the SSC.

5.2 Fire Assessment

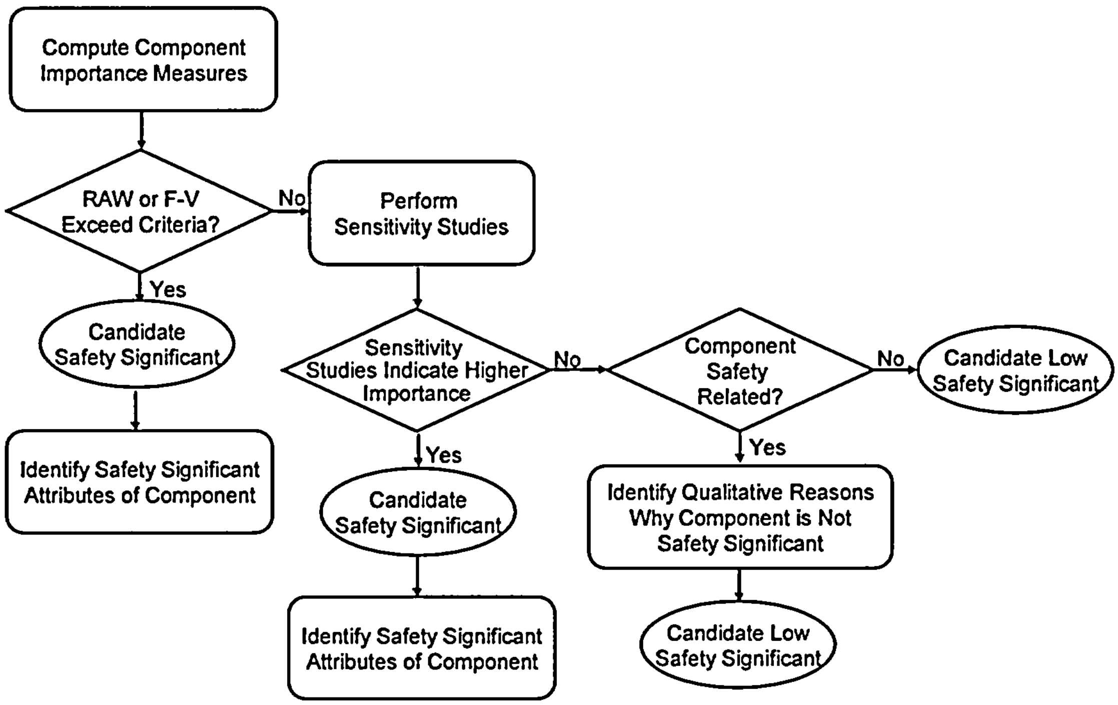
The fire safety significance process takes one of two forms. For plants with a fire PRA, the process is similar to that described for an internal events PRA. This process is shown on Figure 5-3, and is discussed below. Plants that relied upon a FIVE analysis to assess fire risks for the IPEEE should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

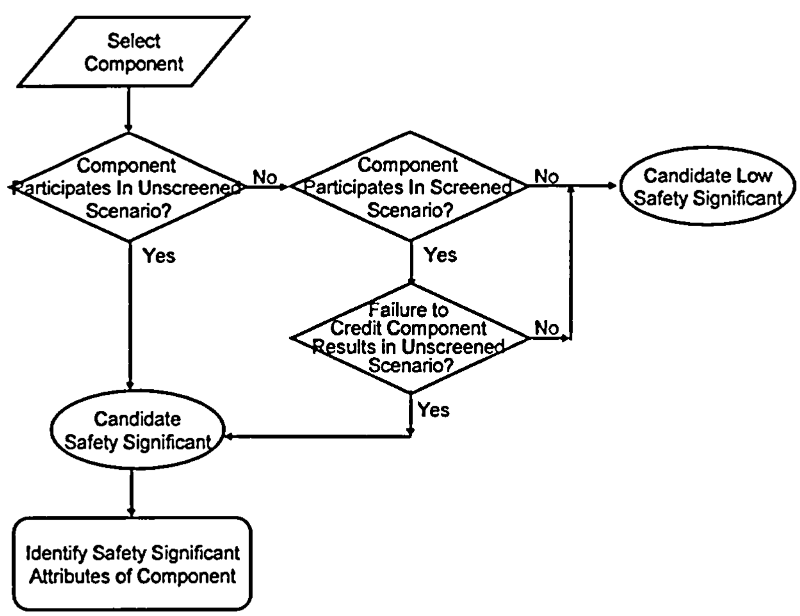
The recommended sensitivity studies for fire PRA are identified in Table 5-3.

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



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. This is conservative since the screening process used in FIVE does not generate numerical estimates of core damage frequency values. However, the option always exists for the licensee to perform a fire PRA to remove this conservatism.

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 SSC 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 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:

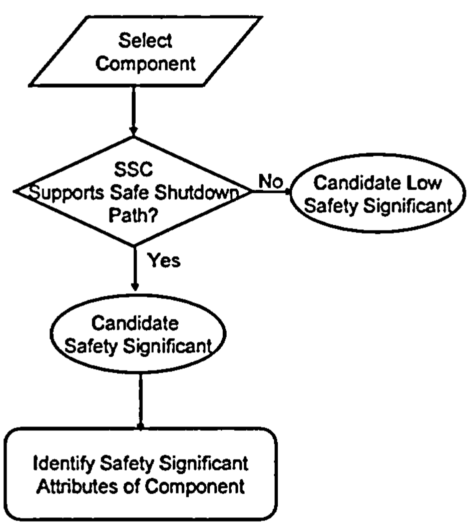
Table 5-4
Sensitivity Studies For Seismic PRA

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



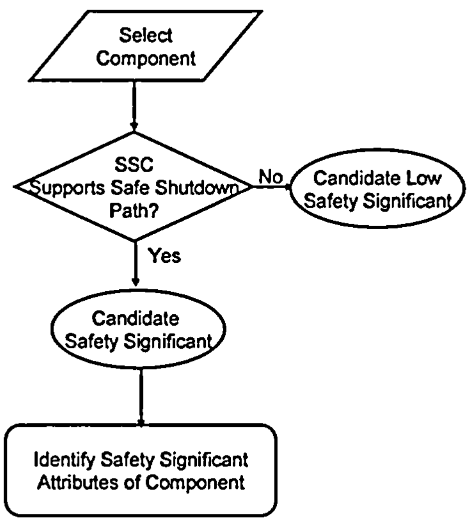
In this process, after identifying the design basis and severe accident functions of the component, the seismic margins analysis is reviewed to determine if the component is credited as part of the safe shutdown paths evaluated. If a component is credited, it is considered safety significant. This is 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 that 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 safe 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) Determine if the SSC is credited as part of the identified safe shutdown path. If a component is credited, it is considered safety significant. The SRP on the NUREG-1407 analysis can be used as guidance in this determination.
    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.

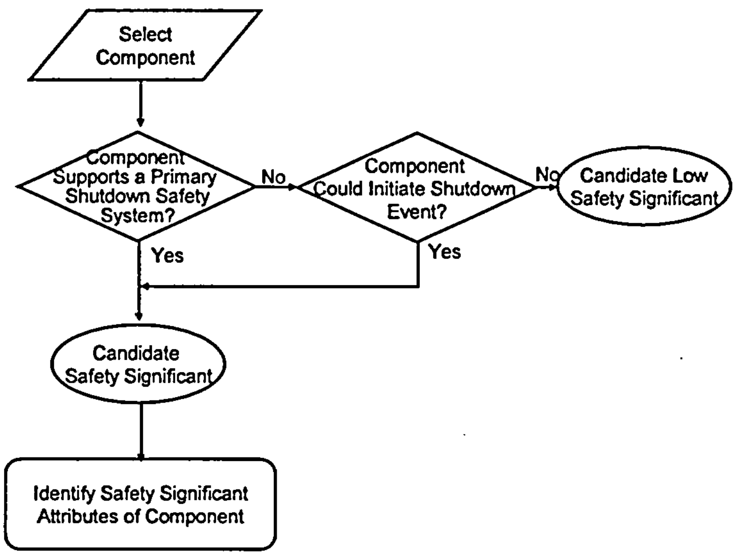
However, if the shutdown PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the shutdown PRA can be considered low safety significant from a shutdown perspective.

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 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 risk 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 internally initiated design basis events considered in the licensee's safety analysis report (i.e., the events that were used to identify the SSC as safety related) 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.

The following process is used in applying Figure 6-1. For each active component function categorized as low risk significant,

- Identify the design basis events that the function is required for.
- For each design basis event, identify the other systems and trains that can support the function or can provide an alternative success path to avoid core damage.
- For each design basis event, identify which region of Figure 6-1 the plant mitigation capability lies **without** credit for the SSC being classified as low safety significant and any identical, redundant SSCs within the system also classified as low safety significant.
- If the result is in the region entitled "Low Safety Significance Confirmed", then the low safety significance of the SSC has been confirmed for that function.
- If the result is in the region entitled "Potentially Safety Significant", then the SSC should be classified as safety significant for the IDP.

When complete, if all SSC functions are confirmed as low safety significant, then the SSC remains Candidate Low Safety Significant for the IDP.

For example, if a BWR found that the low pressure core spray (LPCS) system pumps were low safety significant in the categorization process using risk information, then their categorization would be confirmed using Figure 6-1. In this case, the LPCS pumps have the function of providing coolant makeup to the RPV at low pressure. This function is required either (a) in response to a large LOCA, or (b) in response to other transients and LOCAs where other coolant makeup systems are failed.

For mitigation of a large LOCA, the low pressure coolant injection (LPCI) function of the RHR system can also support the coolant inventory makeup function. The LPCI function is automatic and consists of at least two redundant trains. Thus, for this LOCA event, in the bottom row of Figure 6-1, the presence LPCI as a redundant automatic system confirms the low safety significance of LPCS.

In order to confirm low safety significance in high frequency transient events, such as reactor trip, either two automatic redundant systems are required or 3 or more trains must exist. At BWRs there are multiple coolant inventory makeup systems that could be used without crediting LPCS (i.e., HPCI, RCIC, main feedwater, condensate, and LPCI with ADS). This exceeds the redundancy and diversity requirements for mitigation of these events.

In order to confirm low safety significance for mitigation of a stuck open relief valve, one train plus one redundant system is required. In this case, BWRs have LPCI with ADS and HPCI plus CRD to provide success paths. This provides a redundant system (LPCI/ADS) and one additional diverse train (HPCI/CRD).

In order to confirm low safety significance for mitigation of loss of one safety related DC bus, at least two diverse trains are required. In this case, BWRs would have one train of LCPI and either HPCI (a one train system) or RCIC (a one train system) available to meet the requirement for two diverse trains.

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). Containment defense-in-depth is also assessed for SSCs that play a role in preventing lare containment failures (e.g., due to loss of containment heat removal). For each SSC function categorized as candidate low safety significant, its defense-in-depth is assessed using the following criteria:

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				

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 support a system function that is not considered in CDF and LERF, but would be the only means for preserving long-term containment integrity post-core damage (i.e., containment heat removal)?

In cases where the answer to any of the above questions is "yes," the SSC should be categorized as candidate safety significant. If all of the above questions are answered "no," then low safety significance is confirmed. When complete, if all SSC functions are confirmed as low safety significant, then the SSC remains Condidate Low Safety Significant for the IDP.

In cases where SSCs are identified as safety significant, the safety significant attributes should be defined. 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.

7 PRELIMINARY ENGINEERING CATEGORIZATION OF FUNCTIONS

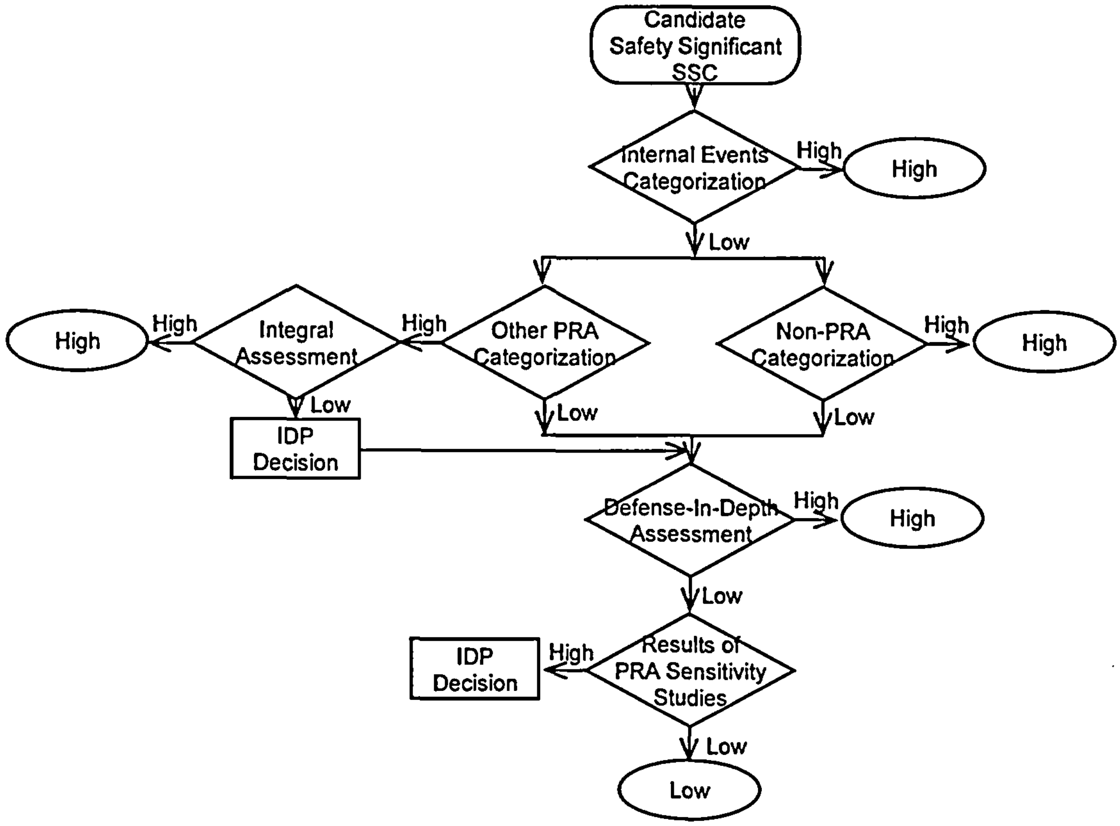
7.1 Engineering Categorization

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. The overall process used in integrating the various categorization inputs is depicted in Figure 7-1.

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 are 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 or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports.

Figure 7-1
Overview of Process for Assigning Overall Safety Significance



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 7-2 provides an example, conceptual layout of the information that summarizes the results and insights that were generated in the categorization process and could be useful for the IDP. This format is for the purposes of identifying the key information that should be communicated to the IDP for use in their decision process. It is expected that additional information will be available at the IDP session that documents the basis for the summary example in the Figure 7-2.

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

- System name
- The function(s) evaluated and the SSCs supporting those functions.
- 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 7-2
EXAMPLE RISK-INFORMED SSC ASSESSMENT WORKSHEET
(FUNCTIONAL BASIS)

System: ______________ Function: ____

Associated Components (or Flowpath): ____

Function Evaluated for Risk? _______ Yes _______ No

SSCs Modeled (explicitly or implicitly) in Risk Assessments: ________________________________

Significance Based on Probabilistic Risk Assessment Tools			
		Potential Risk Significance (High or Low)	Basis for Risk Significance (Include RAW and F-V values where applicable)
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 Individual Sensitivity Studies		
	Change in Risk Significance?	Summary of Findings (Include Delta CDF and LERF or RAW and F-V values where applicable)
Human Error Rates		
Common Cause Failure		
Maintenance Unavailability		
Common Cause Failure		
Others		

Insights From Cumulative Sensitivity Study for the System: ________________________________

Defense-in-Depth Assessment: ________________________________

Categorization in Other Risk Informed Applications (Maintenance Rule, ISI, etc): ________________

<u>Recommended Categorization for Function</u>:

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, because one of the guiding principles of this process is that changes in treatment should not significantly degrade performance for RISC-3 SSCs and should maintain or improve the performance of RISC-2 SSCs, 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. It is not necessary to address the cumulative impact of SSCs for hazards where screening tools such as SMA were used because if they are included in the screening analysis they are considered high safety significant, thus there would be no change in treatment and no change in performance. For categorizations that rely on PRAs, this sensitivity 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. In the extreme, individual components could 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 components 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 that could be impacted by the changes in special treatment.

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.

In cases where the categorization process identifies beyond design basis functions that will be addressed for RISC-1, reducing the unreliability of these safety significant SSCs

by a similar factor may be called for, depending upon the specific changes in special treatment. The cumulative 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., 0.0025) for a re-evaluation of SSCs risk ranking. This may result in re-categorizing 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 risk information 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. This training is for IDP familiarity (i.e., it is not intended to make the IDP PRA "experts").

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, including a list of exempted regulations for low safety significant SSCs,

- The categorization process (e.g., a brief description of Figure 2-1),

- The risk-informed defense-in-depth philosophy and criteria to maintain this philosophy,

- PRA fundamentals,

- Details of the plant-specific PRA analyses that are relied upon for the preliminary categorization, including

  - the modeling scope and assumptions,
  - interpretation of risk importance measures, and
  - the role of sensitivity studies and change in risk evaluations

- The IDP process, including roles and responsibilities.

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, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS.

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.

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

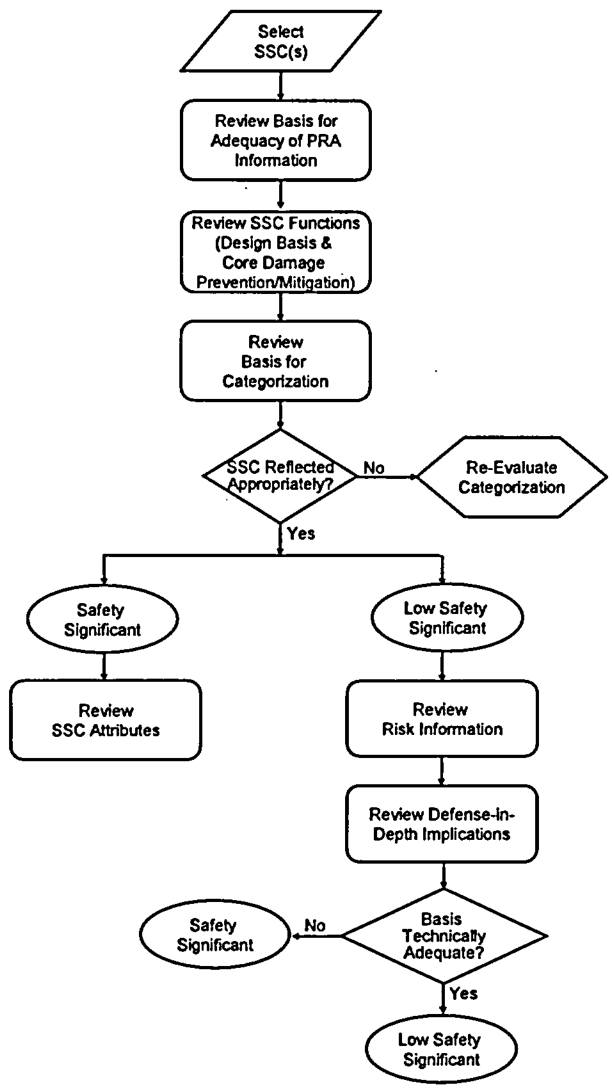
- The designated chairman, panel members, and panel alternates;
- Required training and qualifications for the chairman, members, and alternates;
- Requirements for a quorum, attendance records, agendas, and meeting minutes;
- The decision-making process;
- Documentation and resolution of differing opinions; 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 risk information incorporate those functions. If the IDP determines that the function has not been appropriately reflected, then it is returned to the preliminary categorization process to be 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, then the IDP cannot move that SSC to a low safety significant category. For safety significant 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.

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 been identified as candidate low safety significant, the IDP should review the results to determine whether these functions/SSCs are not implicitly depended upon for risk-significant functions. The IDP should consider whether:

- 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 function/SSC will not compromise the reactor coolant pressure boundary or containment integrity.

- Failure of the associated SSC(s) will fail a safety significant 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).). "Function" here is considered to be one of the "high level" general mitigation categories such as "reactivity control", "hi pressure RPV injection from all sources", etc. That is, the IDP reviews the impact of loss of the SSC against the defense-in-depth remaining to perform the function.

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

- 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).

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 may not be adequate if

- The overall redundancy and diversity among the plant's systems and barriers is not sufficient to ensure that no significant increase in risk would occur;

- Reasonable balance is not 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 not 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 an over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design; and

- Potential for common cause failures is not taken into account in the risk analysis categorization.

If any of the above conditions for either the risk information or the defense-in-depth implicdations are true, low safety significance can still be assigned, if the following condition is met:

- Historical data show that these failure modes are unlikely to occur,
  and

- Such failure modes can be detected and mitigated in a timely fashion, or
- Condition monitoring – leading indicators

Review Safety Margin Implications

Because the only requirements that are relaxed for low safety significant SSCs are those related to treatment, existing safety margins for SSCs arising from the design technical and functional requirements would remain. It is also required that there be reasonable confidence that any potential increases in CDF and LERF be small from assumed changes in reliability resulting from the treatment changes permitted by 50.69. As a result, individual SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results. Therefore, it can be concluded that the sufficient safety margins are preserved. Consequently, no specific assessment of safety margin is required by the IDP.

Review of LSS SSCs

The functions/SSCs initially categorized as LSS may 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 supports 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, RISC categorizing all SSCs within a system or subsystem according to whether the system or subsystem as a whole performs a risk significant function (Section 10.1). 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.

The two options can be explained in more detail as:

1) Assignment of all SSCs in the flow path represented by the function to the safety significance classification of that function. While this is a conservative assignment, it may best suit the cost-benefit assessment for 50.69 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 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 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, but is not limited to:

- 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 that is not a significant flow diversion path, SSCs downstream of the first 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.).

<u>IDP Review of RISC 3 and RISC-4 Components</u>

For SSCs that retain the categorization of the function that they support, no IDP review should be required; there should be no differences from the assessments considered in the initial IDP. For SSCs that are re-categorized to a lower classification (e.g., components in a safety significant function that are determined to be low safety significant based on the above considerations), the new categorization and its basis should be presented to another session of the IDP. In this follow-up session, the IDP would be expected to review the basis for the re-categorization and to assess the impact of this re-categorization on the risk importance and defense in depth implications using the same criteria as in the original IDP session for candidate low safety significant SSCs.

11 PROGRAM DOCUMENTATION AND CHANGE CONTROL

10 CFR 50.69(f) includes requirements for program documentation, change control and records. In general, the implementation of 10 CFR 50.69 can be divided into two phases: 1) the initial implementation that includes the categorization of SSCs and the application of treatment based on that categorization; and 2) the control of changes to the plant that may impact those SSCs or their categorization basis following the initial implementation. This section provides guidance on meeting the requirements of 10 CFR 50.69(f) for these two phases.

11.1 Initial Implementation

The rule requires the licensee or applicant to document the basis for categorization of any SSCs subjected to the categorization process. The heart of this documentation is the procedure used to conduct the categorization process, and a concise summary of the results of the process. For RISC-1 and RISC-2 SSCs, the documentation should include information on any applicable safety-significant beyond design basis functions that were identified. This information is important to the control of any subsequent changes affecting these SSCs following initial implementation. For RISC-3 and RISC-4 SSCs this information should include the basis for concluding that the SSC is low safety significant.

For the purposes of this guidance, initial implementation refers to the first application of the 50.69 rule to a particular system. This may be at the time the first system(s) are categorized under 50.69 or it may be at later time if the licensee chooses a phased approach to categorization wherein only a few systems are categorized each year, for several years.

The rule requires the licensee or applicant to update the FSAR in accordance with 10 CFR 50.71(e) to reflect which systems have been categorized. Following NRC approval to implement 10 CFR 50.69, any changes to the FSAR that reflect alternative treatment of categorized systems should be captured in the licensee's FSAR update process. NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, provides ample guidance on implementing the update process. Any changes to the FSAR associated with initial implementation need not include a supporting review or evaluation under 10 CFR 50.59.

Initial implementation may entail changes to the licensee's quality assurance plan to reflect alternative treatment for categorized systems. Any changes to the quality assurance plan associated with initial implementation need not include a supporting review under 10 CFR 50.54(a). In addition, any regulatory commitments associated with the special treatment requirements in 10 CFR 50.69(b)(1) for SSCs categorized as RISC-3 are no longer applicable to these SSCs and may be dropped at the licensee's discretion.

The waiver of supporting reviews under 10 CFR 50.59 and 10 CFR 50.54(a) is only applicable to the initial implementation of 10 CFR 50.69, i.e., for changes in treatment to

SSCs based on the results of the categorization process. Any other changes to these SSCs are subject to the applicable change control requirements.

11.2 Following Initial Implementation

Subsequent to initial implementation, any changes to alternative treatment for categorized SSCs are subject to applicable change control requirements, e.g., 10 CFR 50.59 and 10 CFR 50.54(a), and must continue to meet the alternative treatment requirements in 10 CFR 50.69.

Changes to categorized SSCs not associated with treatment continue to be governed by the same applicable change control requirements. For RISC-1 and RISC-2 SSCs that have safety-significant beyond design bases functions, the licensee must also maintain reasonable assurance that these functions will be satisfied following the change.

The periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes.

For example, if new information results in a change in categorization of an SSC from RISC-3 to RISC-1, the licensee must reestablish the level of assurance consistent with its safety-significant treatment program that meets the applicable special treatment requirements.

12 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 or the defense in depth implications considerations in previous categorizations have been changed as a result of these later categorization activities. If such changes are found, they should be presented to the IDP for consideration in their deliberations on the categorization of the latest system.

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 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 review of the importance measures used for screening in the categorization process[4].

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

[4] 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

In most cases, the categorization would be expected to be unaffected by changes in the plant-specific PRA. However, in some instances, an updated PRA could result in new RAW and F-V importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. In these cases, the assessment of whether a change in categorization is appropriate should be based on the absolute value of the importance measures. The absolute importance is the product of the base CDF/LERF and the importance measure ([RAW-1] or Fussell-Vesely). This is done in order to not inadvertently assess an SSCs as safety significant when its relative importance (FV and RAW) has gone up, but only due to a decrease in overall CDF & LERF. In cases where the importance measures are different between a prior categorization and an updated result, the categorization reassessments of SSCs that have been previously categorized should be based on the following table:

Table 12-1
IMPACT OF PRA UPDATES ON CATEGORIZATION

Prior Categorization	**Updated CDF/LERF**	**Updated Significance Based on Importance**	**Updated Absolute Importance**	**Updated 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

When a change to the categorization of an SSC is suggested either by a change in plant design or operation that would prevent a safety-significant function from being satisfied or by a change in the PRA model as determined from the absolute importance measures, they should be presented to the IDP for concurrence. In these cases, the IDP would assess the basis for the re-categorization by:

- Review of the primary technical bases for the initial categorization, including the system function(s), the risk importance and the basis for their original categorization,
- Review of the technical basis for the change (in palnt deisgn and operation of PRA model) that has resulted in a suggested change to the SSC categorization including the appropriateness of the manner in which the SSC has been reflected as a result of the change, and
- Review of the new risk importance and defense in depth implications.

The IDP has the final decision regarding the suggested re-categorization.

13 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.

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

Plant-specific Risk Information – Plant-specific evaluations of beyond design basis capability used in the categorization process including PRAs, FIVE, seismic margins assessments, shutdown safety assessments, etc.

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

<u>Table of Contents</u>

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.***. *Licensee's name and unit number,* takes exception to NEI 00-04 and Reg. Guide1.*** 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*

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.***, this submittal, as a risk-informed application, meets the intent and principles of Regulatory Guide 1.174 as described below:

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

- <u>The Proposed Change Is Consistent With The Defense-In-Depth Philosophy</u> – 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.

- <u>The Proposed Change Maintains Sufficient Safety Margins</u> – 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.

- <u>Any Increases in Core Damage Frequency or Risk Should Be Small and Consistent With the Intent of the Commission's Safety Goal Policy Statement</u> – 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.

- <u>The Impact Of The Proposed Change Should Be Monitored Using Performance Measurement Strategies</u> – 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.*** 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 risk information (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.*
- *Site specific seismic hazard curve.*

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 the life of the facility These records will be maintained consistent with the *Subject Plant's* configuration control and document management procedure(s) ***X. *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.***, "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.*** 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

NEI 00-04 (DRAFT - Revision D)

10 CFR 50.69 SSC Categorization Guideline

NUCLEAR ENERGY INSTITUTE

October 2003

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ACKNOWLEDGMENTS

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

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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 with the 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.

The categorization process described in this section is one acceptable way to undertake the categorization of SSCs. Other methods using a different combination of probabilistic and deterministic approaches and criteria can be envisioned. However, it is expected that the guiding principles (Section 1.3) of this guidance would be maintained. Licensees wishing to use a different method for categorizing SSCs using risk-informed insights need to submit the methodology for NRC review and approval.

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[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.

Changes to this guideline are controlled through the normal regulatory change control processes. Section 11 provides guidance on program documentation and change control.

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 could occur, 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.

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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

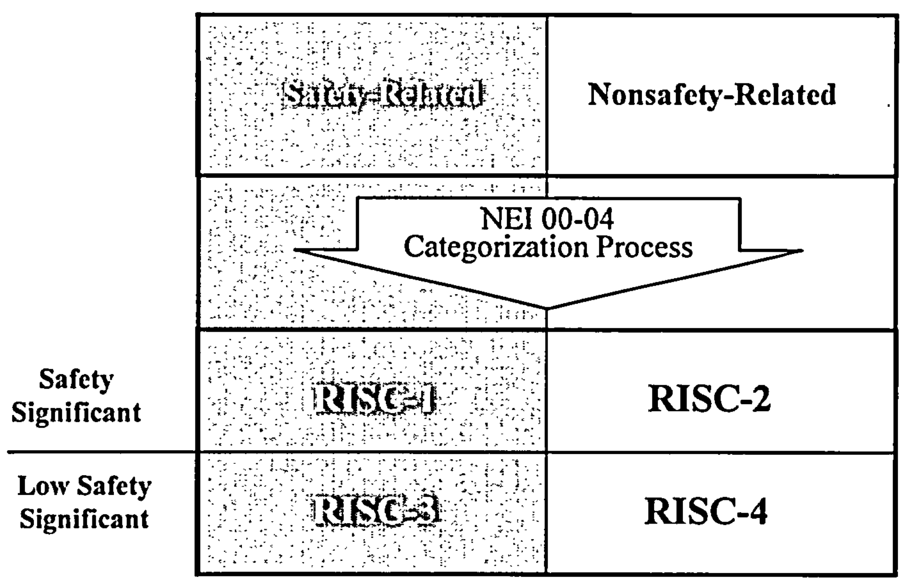
resources on equipment that has safety significance. This guideline addresses the use of risk insights to define the scope of equipment that should be subject to NRC special treatment provisions as defined in §50.69.

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 augmented 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 safety related SSC will be re-categorized as RISC-1 unless a basis can be developed for re-categorizing it as RISC-3.
- Attribute(s) that make a SSC safety-significant should be documented.

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1.4 VOLUNTARY AND SELECTIVE IMPLEMENTATION

US nuclear generating plants have attained and maintained an outstanding safety performance record. The existing NRC regulations together with the NRC's regulatory oversight and inspection processes clearly provide adequate protection of public health and safety. As a result, the decision to 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.

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1.5 CATEGORIZATION PROCESS SUMMARY

The NEI 00-04 categorization process embodies the principles of risk-informed regulation described in Reg. Guide 1.174 (Figure 1-2). The plant-specific risk analyses provide an initial input to the process. SSCs identified as high safety significant (HSS) by the risk characterization process are identified for an integrated decision-making panel (IDP). The IDP cannot re-categorize an SSC identified by the risk analysis as HSS. The IDP function is to review the assessment and assure that the system functions and operating experience have been appropriately considered in the risk analyses.

SSCs that are safety related and considered to be low safety significant (LSS) based on the plant-specific risk analyses are evaluated in a defense-in-depth characterization process. This deterministic process addresses the role of the SSC with respect to both core damage prevention and containment performance. If defense-in-depth characterization identifies that the SSC should be considered HSS, then it is re-categorized as HSS and recommended to the IDP as a RISC-1 SSC. Here again, the IDP cannot re-categorize an SSC identified by the as HSS. The IDP function is to review the assessment and assure that the system functions and operating experience have been appropriately considered.

If an SSC is found to be LSS by both the risk categorization process and the defense-in-depth characterization process, then it is recommended to the IDP to be LSS. The IDP reviews the categorization process applied to the SSC and, if the IDP feels that the operating experience or functions merit a HSS categorization, they can re-categorize it.

Thus, only if an SSC is found to be of low safety significance by all three (i.e. the risk characterization process, the defense-in-depth characterization process and IDP review), will it be categorized as low safety significant.

Risk Characterization

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

- Internal Event Risks
- Fire Risks
- Seismic Risks
- Other External Risks (e.g., tornados, external floods, etc.)

- Shutdown Risks

Separate evaluation is appropriate to avoid reliance on a combined result that fails to address these differences.

Table 1-1 provides a summary of the alternative approaches taken to address each risk contributor. A brief description of each of these aspects is described.

Internal Event Risks

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A high quality PRA is required for the categorization of SSCs relative to internal events, at-power risks. Importance measures related to Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) are used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2). In addition, several sensitivity studies are defined which exercise key areas of uncertainty in the PRA (e.g., human reliability, common cause failures, and no maintenance plant configuration). If an SSC that had been initially identified as low safety significant is found to exceed the safety significance thresholds in a sensitivity study, this information is provided to the IDP, along with an explanation of why the sensitivity study identified the SSC to be safety significant.

Fire Risks

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A fire risk analysis, either a plant-specific fire PRA or a Fire Induced Vulnerability Evaluation (FIVE) analysis that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to fire risks. If a fire PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the fire risk contribution is shown to be sufficiently small (in comparison to the internal events risk) as to make the overall safety significance of the SSC low (RISC-3 or -4) in the Integrated Importance Assessment (see below). Sensitivity studies, including fire-specific sensitivity studies, are also identified and used in a similar manner.

In the event a FIVE analysis is used, the categorization process is necessarily more conservative (i.e., designed to identify more SSCs as safety significant). This is due to the fact that FIVE is a screening tool. As such, the resulting scenarios and frequencies have an uneven level of realism. Thus, importance measures are not an effective means for identifying safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the mitigation of any unscreened fire scenario (i.e., retained for consideration in the FIVE analysis) as safety significant. In addition, all screened scenarios are reviewed to identify any system functions and associated SSCs that would result in a scenario being unscreened, if that system function was not credited. This measure of safety significance assures that the SSCs that were required to maintain low fire risk are retained as safety significant.

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Figure 1-2
Summary of NEI 00-04 Categorization Process

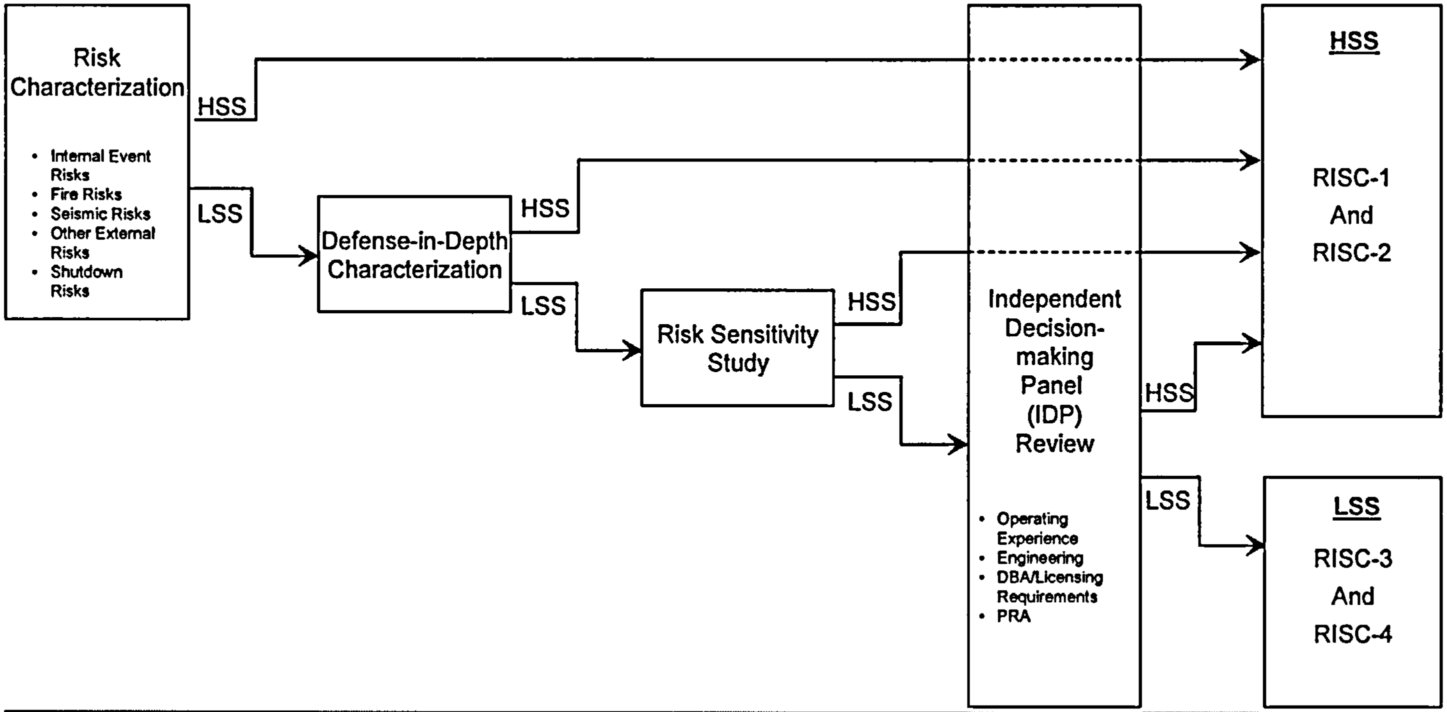

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Table 1-1
Summary of Risk Significance Characterization Used in NEI 00-04

Risk Source	Alternative Approaches	Scope of Safety Significant SSCs
Internal Events	PRA Required	Per PRA Risk Ranking
	Screening Approaches Not Allowed	n/a
Fire	Fire PRA	Per PRA Risk Ranking
	FIVE (Fire Induced Vulnerability Evaluation)	All SSCs Necessary to Maintain Low Risk
Seismic	Seismic PRA	Per PRA Risk Ranking
	SMA (Seismic Margins Analysis)	All SSCs Necessary to Maintain Low Risk
High Winds, External Floods, etc.	PRA	Per PRA Risk Ranking
	IPEEE Screening	All SSCs Necessary to Protect Against Hazard
Shutdown	Shutdown PRA	Per PRA Risk Ranking
	Shutdown Safety Plan	All SSCs Required to Support Shutdown Safety Plan

Seismic Risks

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A seismic risk analysis, either a plant-specific seismic PRA or a seismic margin analysis (SMA) that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to seismic risks. If a seismic PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the seismic risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (RISC-3 or -4) using the integrated importance assessment. Sensitivity studies, including seismic-specific sensitivity studies, are also identified and used in a similar manner.

In the event an SMA is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant). This is due to the fact that SMA is a screening tool. As a screening tool, importance measures are not available to identify safety significance. The NEI 00-04 approach identifies all system functions and associated SSCs that are involved in the seismic margin success paths as safety significant. This measure of safety significance assures that the SSCs that were required to maintain low seismic risk are retained as safety significant. The seismic PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but the SMA identifies them as safety significant regardless of their capacity, frequency of challenge or level of functional diversity.

Other External Risks

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For other external event risks, either a plant-specific external event PRA or a screening analysis that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to other external risks. If an external hazard PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the other external hazard risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (see integrated importance assessment below). Sensitivity studies are also identified and used in a similar manner.

In the event an screening analysis is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant). The NEI 00-04 approach identifies all system/structure functions and associated SSCs that are involved in protecting against the external hazard as safety significant. An example might be a tornado missile barrier. Using a PRA, some barriers might be found to be of low safety significance, depending on the site-specific frequency of tornadoes and the equipment protected by the barrier. Using a screening method, the barrier would be identified as safety significant without regard to those other factors. This measure of safety significance is much more restrictive than the importance measures used in the external hazard PRA and would be expected to yield a larger set of safety significant SSCs than the external hazard PRA. The PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but

the screening approach identifies them as safety significant regardless of their capacity, frequency of challenge or level of functional diversity.

Shutdown Risks

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A shutdown risk analysis, either a plant-specific shutdown PRA or a shutdown safety management plan that reflects the current as-built, as-operated plant is used to identify SSCs that are safety significant due to shutdown risks. If a shutdown PRA is available, then importance measures are once again used to identify the safety significant functions and all SSCs that support those functions are categorized as safety significant (RISC-1 or -2), unless the shutdown risk contribution is shown to be sufficiently small as to make the overall safety significance of the SSC low (see integrated importance assessment below). Sensitivity studies, including shutdown-specific sensitivity studies, are also identified and used in a similar manner.

In the event a shutdown safety management plan is used, the categorization process is, once again, more conservative (i.e., designed to identify more SSCs as safety significant) than a plant specific PRA. This is due to the fact that the shutdown safety management plan provides safety function defense in depth without regard to the likelihood of demand or reliability of the functions credited. The NEI 00-04 approach identifies all SSCs necessary to support primary shutdown safety systems as safety significant. This measure of safety significance assures that the SSCs that were required to maintain low shutdown risk are retained as safety significant. The shutdown PRA credits all of the same SSCs in a probabilistic framework so some may avoid being identified as safety significant using the PRA, but the shutdown safety management plan approach identifies them as safety significant regardless of the frequency of challenge or level of functional diversity.

Integrated Importance Assessment

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Each risk contributor is initially evaluated separately due to the significant differences in the methods, assumptions, conservatisms and uncertainties associated with the risk evaluation of each. In general, the quantification of risks due to external events and non-power operations tend to contain more conservatisms than internal events, at-power risks. As a result, performing the categorization simply on the basis of a mathematically combined total CDF/LERF would lead to inappropriate conclusions. However, it is desirable in a risk-informed process to understand safety significance from an overall perspective, especially for SSCs that were found to be safety significant due to one or more of these risk contributors.

In order to facilitate an overall assessment of the risk significance of SSCs, an integrated computation is performed using the available importance measures. This integrated importance measure essentially creates a weighted-average importance based on the importance measures and the risk contributed by each hazard (e.g., internal events, fire, seismic PRAs). The weighted importance measures can be significantly influenced by the relative contribution of the hazard. For example, an SSC that is very important for a

hazard that contributes only 1% to the total CDF/LERF would be found to have very low importance measures when the integrated assessment is performed. In no case will the integrated importance measure be larger than the largest of the individual hazard importance measure. This integrated assessment allows the IDP to determine whether the safety significance of the SSC should be based on the significance for that individual hazard or from the overall integrated result, avoiding a strict reliance on a mathematical formula that ignores the significant dissimilarities in the calculated risk results.

Defense in Depth Characterization

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For safety related SSCs initially identified as low safety significant (RISC-3) from the results of the risk significance categorization, an additional defense-in-depth assessment is performed. The defense in depth assessment is based on a set of deterministic criteria based on design basis accident considerations to assure that adequate redundancy and diversity will be retained. This assessment evaluates the SSC functions with respect to core damage mitigation, early containment failure/bypass, and long term containment integrity. If one of these SSC functions is found to be safety significant with respect to defense-in-depth, then it is considered safety significant and re-categorized as safety significant (RISC-1) for presentation to the IDP.

Risk Sensitivity Study

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The final step in the process of categorizing SSCs into risk-informed safety classifications involves the evaluation of the risk implications of changes in special treatment. This risk sensitivity study is performed using the available PRAs to evaluate the potential impact on CDF and LERF, based on a postulated change in reliability. In this risk sensitivity study, the unreliability of all low safety significant SSCs is increased simultaneously by a common multiplier as an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of all low safety significant SSCs. A simultaneous degradation of all SSCs 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 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. 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, actual net increase in risk.

In cases where the licensee does not use a PRA in the categorization process, the sensitivity study remains a viable indication of potential limiting risk increases. This is due to the fact that the categorization processes for hazards that do not have a PRA is done in a manner that assures the risk sensitive SSCs are categorized as safety significant. For example, in the event a seismic margins analysis (SMA) is used for the categorization, all of the SSCs necessary to maintain the current risk levels are considered

safety significant. As a result, there would not be any change in the treatment for the SSCs that are credited in mitigating seismic risk.

Integrated Decision-making Panel Review

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The Integrated Decision-making Panel (IDP) is a multi-discipline panel of experts that reviews the results of the initial categorization and finalizes the categorization of the SSCs/functions. The purpose of the IDP is to assure that the appropriate considerations from plant design and operating practices and experience are reflected in the categorization input.

The IDP considers the safety significance of the SSCs based on:

- the PRA assessments and sensitivity studies,
- a defense in depth assessment from an operational perspective,
- insights from other risk informed programs (e.g., Maintenance Rule, Risk Informed ISI, etc.), and
- operational and maintenance experience.

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In order for an SSC/function to be recommended to the IDP as low safety significant, it must have been identified as low safety significant from the perspective of

- Internal Event Risks
- Fire Risks
- Seismic Risks
- Other External Risks
- Shutdown Risks

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If it is an SSC that is currently safety related, then the defense in depth assessment must also have shown that the SSC is not safety significant. Finally, the risk sensitivity study verifies that the combined impact of a postulated simultaneous degradation in reliability of all low safety significant SSCs would not result in a significant increase in CDF & LERF.

If an SSC is only identified as safety significant based on a non-internal events PRA (and was not found to be significant in the integrated importance assessment), or by one of the mandatory sensitivity studies, then the IDP will be presented the results and will use other knowledge and experience to decide whether the SSC should be safety significant.

The IDP will not over-rule the categorization process to make an SSC/function low safety significant when the process identifies it as safety significant (i.e., will not move it from RISC-1 to RISC-3). The IDP may, however, identify that the SSC/function was not appropriately reflected in engineering assessment which may result in a new categorization, based on a revised evaluation.

Conclusions

The categorization methodology used to define the low safety significant SSCs, as described in NEI-00-04, assures any reduction in component reliability as a result of changes in treatment will have a negligible impact on plant risk. This degree of assurance is provided by a multi-layered approach to identifying the low safety significant SSCs that includes PRA, deterministic assessments and engineering judgment. In addition, two different plant organizational functions (engineering and the IDP) perform assessments from their own unique perspective. In either the engineering or the IDP assessment, if any of these three elements indicates that an SSC is safety significant, then that categorization (safety significant) is assigned.

In terms of the scope of the PRA used in the risk assessment portion of the categorization process, a reasonable degree of confidence that risk significant SSCs will be appropriately identified can be maintained with a quality internal events at-power PRA. Screening assessments for other initiating events and other modes of operation identify the SSCs necessary to maintain low risk.

The number of independent criteria that an SSC must satisfy in order to be categorized as low safety significant provides a high level of assurance that only SSCs that are truly low safety significant will be categorized as such.

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.

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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 risk information to assure that they are adequate to support this application. More detail is provided on this step in Section 3.

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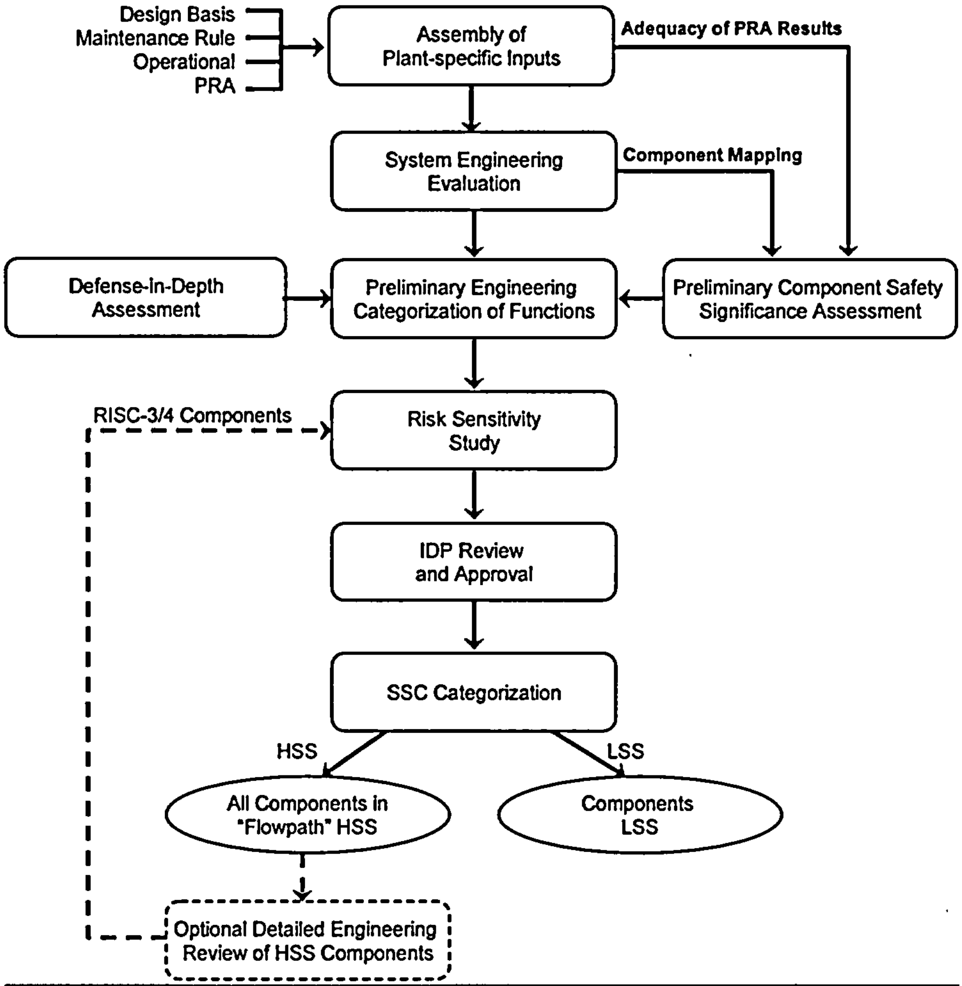
System Engineering Assessment

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 risk information to identify components that are candidate 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.

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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 identifies any HSS SSCs that can be categorized as LSS results of that re-catgegorization are 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.

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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 is 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 Risk 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 satisfy 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.

When risk information is used to provide insights into the integrated decision-making panel, it is expected that the risk information will have been subject to quality measures. The following describes methods acceptable to ensure that the risk information is of sufficient quality to be used for regulatory decisions and meets the quality standards described in Reg. Guide 1.174:

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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.

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- 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 the adequacy of the risk information used in the categorization process (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 risk information can be used to support the categorization process, provided it can be shown that the appropriate quality provisions have been met.

Other 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 Risk Information

Figure 3-1 depicts the approach to be employed in demonstrating the adequacy of risk information used in the categorization of SSCs. The adequacy of the risk information builds upon the efforts to review and evaluate the adequacy of the plant-specific internal event full power PRA. There are two options for demonstrating the adequacy of the results of the internal events PRA for use in the categorization process.

The first approach is to utilize the industry peer review process (NEI 00-02). In a letter dated April 24, 2000, NEI requested the NRC staff review the suitability of the peer review process described in NEI 00–02 to address PRA quality issues for this application. NRC issued a request for additional information on September 19, 2000, to which NEI responded by letter dated January 18, 2001. By letter dated April 2, 2002 (ADAMS accession number ML020930632), the NRC staff sent to NEI draft staff review guidance that was developed as a result of its review of NEI 00–02, for intended use for § 50.69 applications.

The staff review guidance is for a focused review of the plant-specific PRA based on a review of NEI 00–02 and NEI 00–04. In order to reach the conclusion that the PRA results support the proposed categorization, the review guidance is structured to lead the staff reviewer to either look for evidence that the impact of a given peer review issue on

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PRA results has been adequately addressed in the peer review report and, when necessary, has been identified for consideration by the IDP, or to request further information from the licensee.

If a licensee decides to utilize the NEI 00-02 peer certification to demonstrate the adequacy of the PRA results, the staff review guidance would be used to identify and address potential issues prior to use of the PRA.

The second approach would rely upon the process currently described in draft regulatory guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." This guide provides guidance on the NRC position on voluntary consensus standards for PRA (in particular on the ASME standard for internal events PRAs) and industry PRA documents (e.g., NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guideline"). Ultimately, this guide will be modified to address PRA standards on fire, external events, and low power and shutdown modes, as they become available. The NRC has also developed a draft supporting Standard Review Plan, SRP 19.1, to provide guidance to the staff on how to determine whether a PRA providing results being used in a decision is technically adequate.

If a licensee decides to utilize the DG-1122 process to demonstrate the adequacy of the PRA results, it would be used to identify and address potential issues prior to use of the PRA in support of any 50.69 application.

Both processes rely upon peer review findings as a significant measure of the adequacy of the PRA results. All significant peer review findings will 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 risk information used in the categorization process, such as Fire PRAs, FIVE, Seismic PRAs, SMAs 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 risk information to be used should be documented in a characterization of the adequacy of the PRA. This characterization will be provided to the IDP as a basis for the adequacy of the risk information used in the categorization process and will be summarized in the submittal to the NRC. At a minimum, this characterization should include the following:

Deleted: 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 gene ... [1]

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Full Power Internal Events PRA

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- 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, if NEI 00-02 is used, or lower than ASME Capability Category II, if the DG-1122 process is used.
- The disposition of any significant peer review findings.
- Identification of and basis for any sensitivity analyses necessary to address identified findings.
- Considerations identified by the NRC in their letter to NEI [Ref. 15], if the NEI 00-02 process is used.

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Other Risk Information (including other PRAs and screening methods)

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

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The Integrated Decision-making Panel (IDP) should use this information, in combination with the results of the categorization analyses and other information, to finalize the categorization of each function/SSC. The process to be used to justify the adequacy of the risk information is also summarized in the submittal to the NRC.

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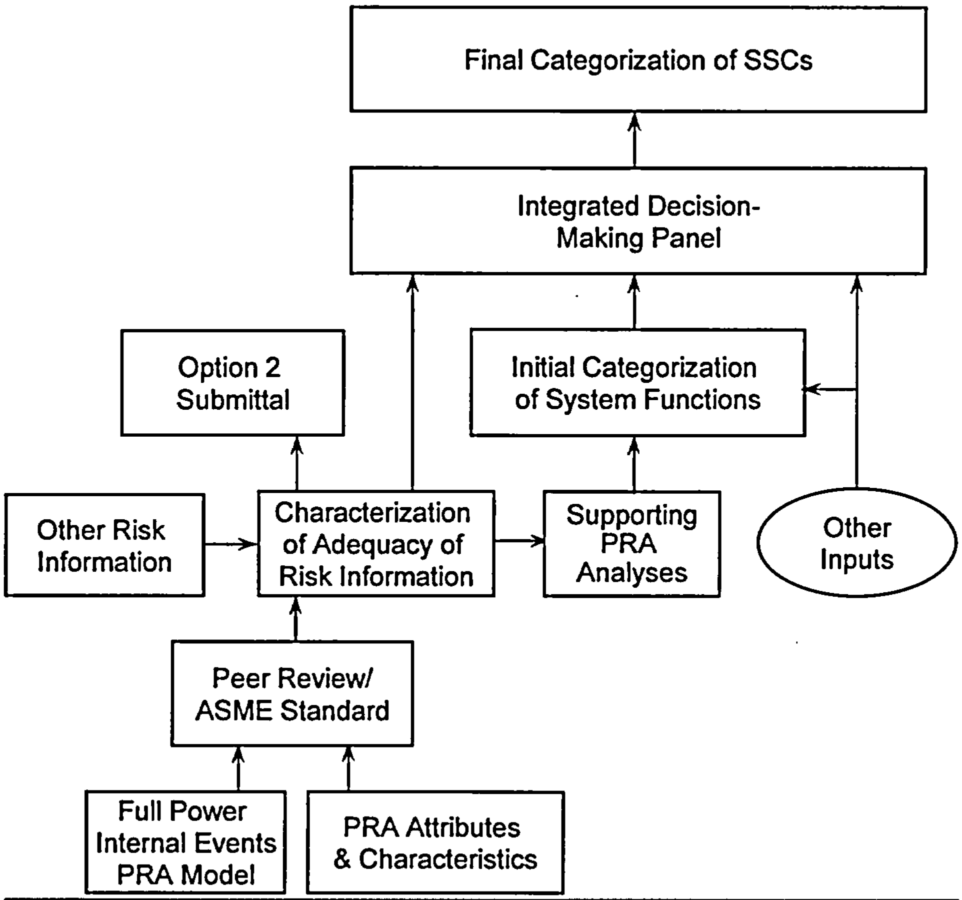
Figure 3-1

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 Considered in the Plant-specific Risk Information
- Operational Functions

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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.

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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-660, *Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities* (Ref. 16), is being implemented at the same time.[2]

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   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.

[2] If this code case is not endorsed at the time of submittal, then the licensee will describe the process to be used in the Option 2 submittal.

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.

Significance from Internal Events

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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 candidate 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 event 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 SSC is categorized as candidate low safety significant from the standpoint of internal event risks. The evaluation is continued with fire risk.

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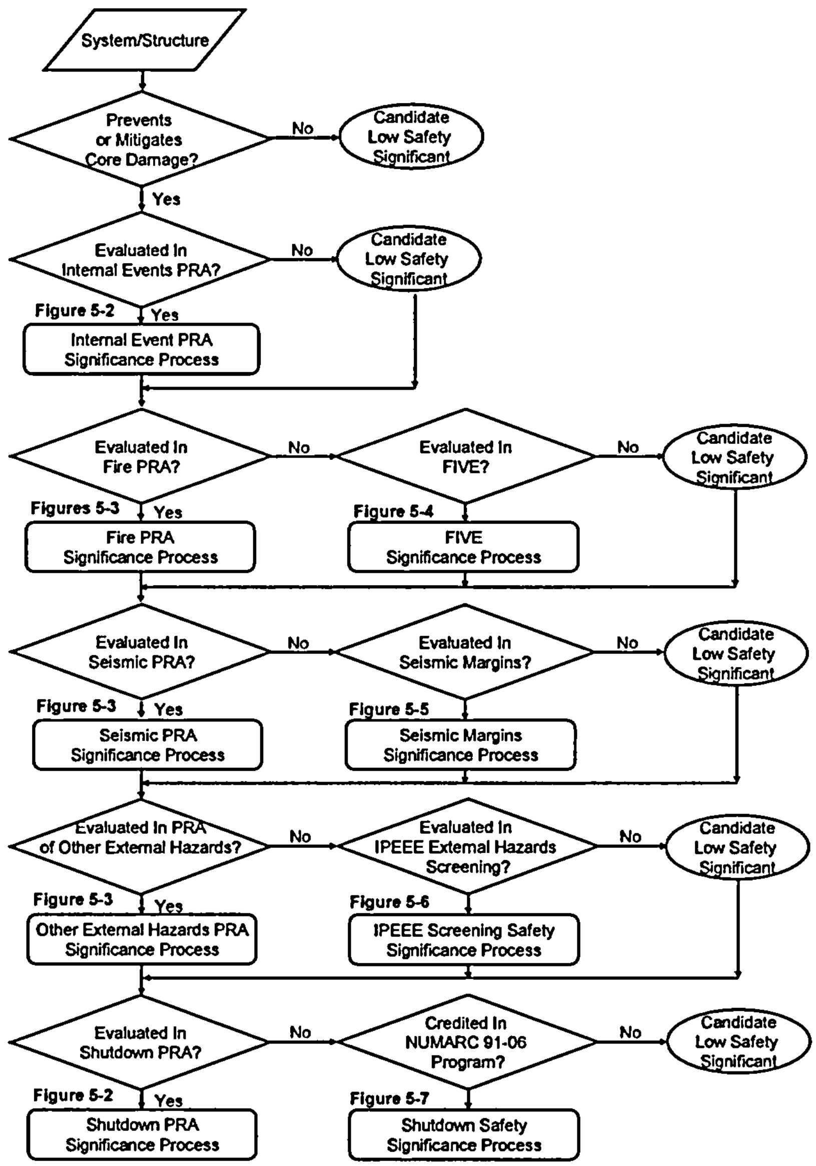
Figure 5-1

USE OF RISK ANALYSES FOR SSC CATEGORIZATION



Significance from Fire Events

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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 they 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 SSC is categorized as candidate low safety significant from the standpoint of fire risks.

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Significance from Seismic Events

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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 SSC is categorized as candidate low safety significant from the standpoint of seismic risk.

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Significance from Other External Events

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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 SSC is categorized as candidate low safety significant from the standpoint of other external risks.

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Significance from Shutdown Events

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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 SSC is categorized as candidate low safety significant from the standpoint of shutdown risk.

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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. For obtaining risk insights from the PRA for passive pressure boundary components, additional guidance is provided in ASME Code Case N-660, 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[3].

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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, because 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 can be expected to be affected by the special treatment requirements being evaluated.

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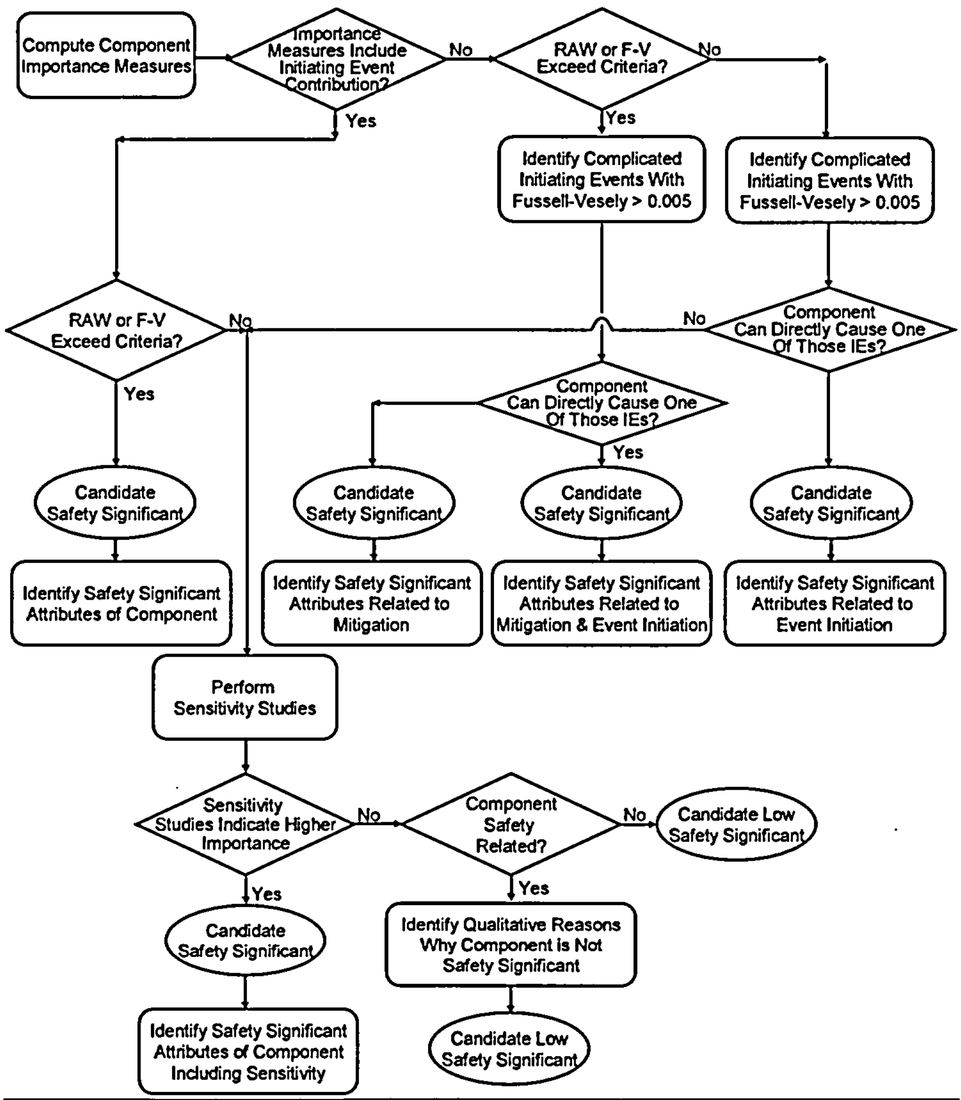
[3] If these code cases and methods are not endorsed at the time of submittal, then the licensee will describe the process to be used in the Option 2 submittal.

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 failure modes of the individual component. In the case of RAW, the common cause event is considered using a different criterion than the individual component RAW. The RAW for common cause events reflects the relative increase in CDF/LERF that would exist if a set of components or an entire system was made unavailable. As a result, the risk significance of the RAW values of common cause basic events are considered separately from the basic events that reflect an individual component. As with the individual component RAW values, if the component being evaluated is included in more than one common cause basic event, the maximum of the common cause RAW values is used to evaluate the significance.

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The importance measure criteria used to identify candidate safety significance are:

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- Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005
- Maximum of component basic event RAW values > 2
- Maximum of applicable common cause basic events RAW values > 20.

If any of these criteria are exceeded it is considered candidate safety significant.

For example, a motor operated valve may have a number of basic events associated with it (e.g., "failure to open" and "failure to close"), 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). Some SSCs perform multiple functions (e.g., circuit breakers can perform a function necessary for pump operation and a function necessary to protect the bus in case of a fault. In these cases, basic events should be mapped to the appropriate functions so that the significant functions can be identified.

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	CCF RAW
1) Valve 'A' Fails to Open	0.002	1.7	n/a
2) Valve 'A' Fails to Remain Closed	0.00002	1.1	n/a
3) Valve 'A' In Maintenance (Closed)	0.0035	1.7	n/a
4) Common Cause Failure of Valves 'A', 'B' & 'C' to Open	0.004	n/a	54
5) Common Cause Failure of Valves 'A' & 'B' to Open	0.0007	n/a	5.6
6) Common Cause Failure of Valves 'A' & 'C' to Open	0.0006	n/a	4.9
Component Importance	0.01082 (sum)	1.7 (max)	54 (max)
Criteria	> 0.005	>2	>20
Candidate Safety Significant?	**Yes**	**No**	**Yes**

In the above example, Valve 'A' would be considered candidate safety significant on two bases, either one would be sufficient to identify the component as candidate safety significant. The total Fussell-Vesely exceeded the criterion of 0.005 and the RAW criterion was also met for the common cause group including Valve 'A'. Thus, both Valve 'A', Valve 'B' and Valve 'C' would be identified as candidate safety significant due to this criterion. The component failure mode which contributes significantly to the importance of Valve 'A' is failure to open (failure modes 1, 4, 5 and 6). 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.

In cases where the internal events core damage frequency is dominated by an internal flooding result that has a conservative bias, it is appropriate to break the evaluation of importance measures into two steps. This prevents the conservative bias of the flooding analysis from masking the importance of SSCs not involved in flood sceanrios. 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 being safety significant. The recommended sensitivity studies for internal events PRA are identified in Table 5-2.

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Deleted: 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 ... [2]
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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, including sensitivity studies, 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. In addition, the truncation level used should be sufficient to identify all the SSCs with RAW>2. 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 is reasonable. In linked fault tree PRAs that do not use pre-

solved cutsets, the truncation limit should be evaluated to ensure that converged importance measures are being used. If the model relies on a pre-solved set of cutsets to calculate CDF, then the RAW values may be underestimated and the nominal truncation level may not be capable of identifying all the RAW>2 SSCs, even in a converged solution. Therefore, the truncation of pre-solved set of cutsets should be checked to ensure that the CDF and LERF solutions are sufficiently adequate by justifying the omitted SSCs with RAW>2. In some cases, this may be best handled by complete re-solution of the model without credit for the SSC.

Deleted: 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 in the scope of this guideline unless the fire risk analysis allows the quantification 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.

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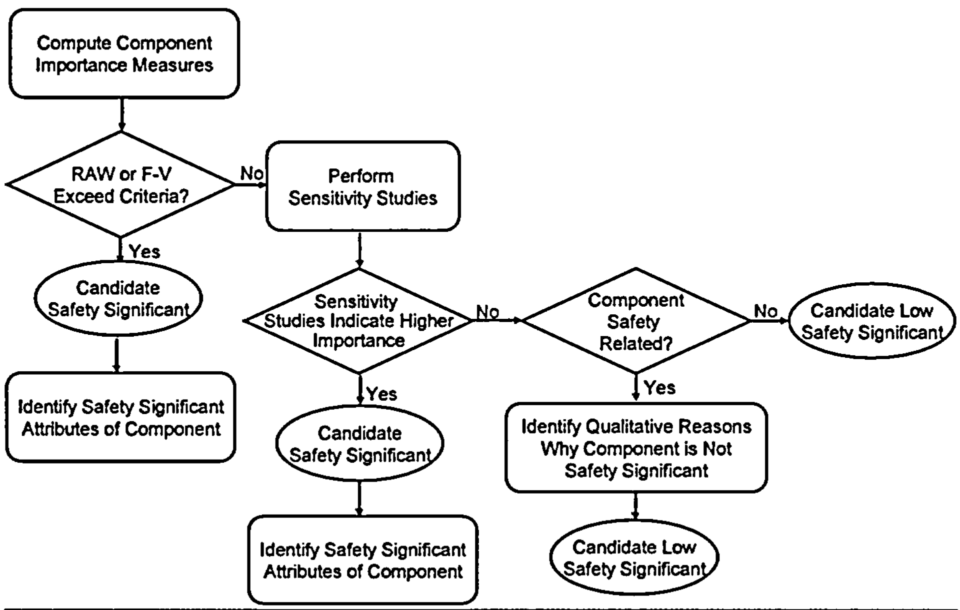
Figure 5-3

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.

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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.

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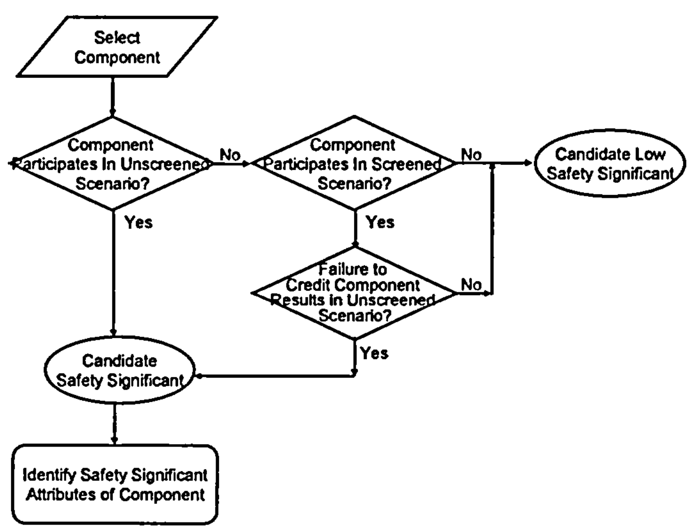
Figure 5-4
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN FIVE



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. This is conservative since the screening process used in FIVE does not generate numerical estimates of core damage frequency values. However, the option always exists for the licensee to perform a fire PRA to remove this conservatism.

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 SSC 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 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 95^{th} percentile value
• Decrease all human error basic events to their 5^{th} 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.

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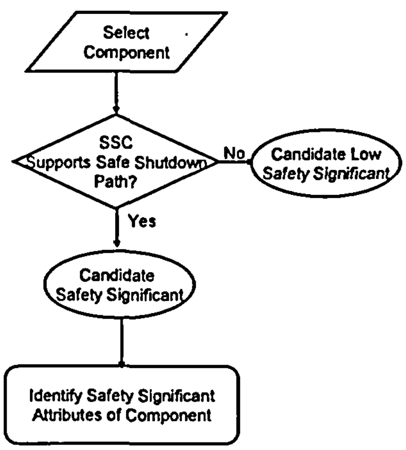
Figure 5-5
SAFETY SIGNIFICANCE PROCESS FOR
SYSTEMS AND COMPONENTS ADDRESSED IN SEISMIC MARGINS



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

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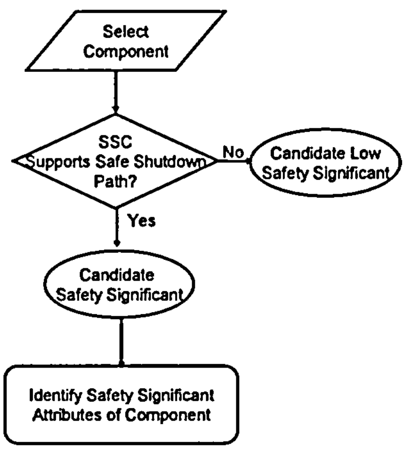

The generalized safety significance process for plants with an external hazard PRA is the same as the process for an internal events PRA. As for seismic risk, the risk importance process is slightly modified to consider the fact that plant components cannot initiate external events such a floods, tornadoes, 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 safe 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) Determine if the SSC is credited as part of the identified safe shutdown path. If a component is credited, it is considered safety significant. The SRP on the NUREG-1407 analysis can be used as guidance in this determination.
   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.

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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.

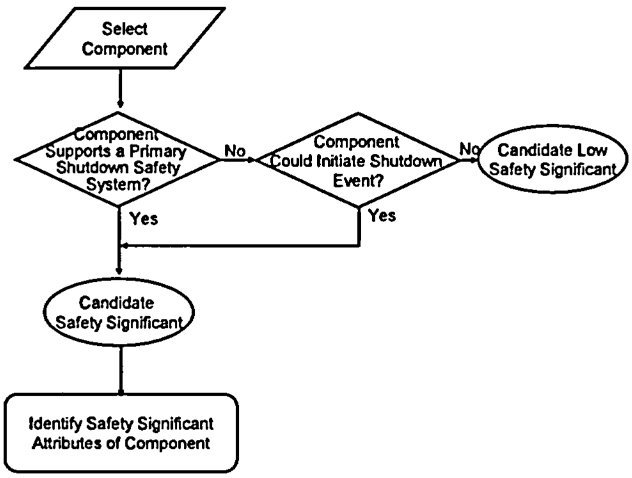
However, if the shutdown PRA CDF is a small fraction of the internal events CDF (i.e., <1%), then safety significance of SSCs considered in the shutdown PRA can be considered low safety significant from a shutdown perspective.

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 margin to fulfill the safety function.
- It does not require extensive manual manipulation to fulfill its safety function.

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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

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

Where,

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

Integrated Risk Achievement Worth Importance

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

Where,

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

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 risk 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.

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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 internally initiated design basis events considered in the licensee's safety analysis report (i.e., the events that were used to identify the SSC as safety related) 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.

The following process is used in applying Figure 6-1. For each active component function categorized as low risk significant,

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- Identify the design basis events that the function is required for.
- For each design basis event, identify the other systems and trains that can support the function or can provide an alternative success path to avoid core damage.
- For each design basis event, identify which region of Figure 6-1 the plant mitigation capability lies without credit for the SSC being classified as low safety significant and any identical, redundant SSCs within the system also classified as low safety significant.
- If the result is in the region entitled "Low Safety Significance Confirmed", then the low safety significance of the SSC has been confirmed for that function.
- If the result is in the region entitled "Potentially Safety Significant", then the SSC should be classified as safety significant for the IDP.

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When complete, if all SSC functions are confirmed as low safety significant, then the SSC remains Candidate Low Safety Significant for the IDP.

For example, if a BWR found that the low pressure core spray (LPCS) system pumps were low safety significant in the categorization process using risk information, then their categorization would be confirmed using Figure 6-1. In this case, the LPCS pumps have the function of providing coolant makeup to the RPV at low pressure. This function is required either (a) in response to a large LOCA, or (b) in response to other transients and LOCAs where other coolant makeup systems are failed.

For mitigation of a large LOCA, the low pressure coolant injection (LPCI) function of the RHR system can also support the coolant inventory makeup function. The LPCI function is automatic and consists of at least two redundant trains. Thus, for this LOCA event, in the bottom row of Figure 6-1, the presence LPCI as a redundant automatic system confirms the low safety significance of LPCS.

In order to confirm low safety significance in high frequency transient events, such as reactor trip, either two automatic redundant systems are required or 3 or more trains must exist. At BWRs there are multiple coolant inventory makeup systems that could be used without crediting LPCS (i.e., HPCI, RCIC, main feedwater, condensate, and LPCI with ADS). This exceeds the redundancy and diversity requirements for mitigation of these events.

In order to confirm low safety significance for mitigation of a stuck open relief valve, one train plus one redundant system is required. In this case, BWRs have LPCI with ADS and HPCI plus CRD to provide success paths. This provides a redundant system (LPCI/ADS) and one additional diverse train (HPCI/CRD).

In order to confirm low safety significance for mitigation of loss of one safety related DC bus, at least two diverse trains are required. In this case, BWRs would have one train of LCPI and either HPCI (a one train system) or RCIC (a one train system) available to meet the requirement for two diverse trains.

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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). Containment defense-in-depth is also assessed for SSCs that play a role in preventing lare containment failures (e.g., due to loss of containment heat removal). For each SSC function categorized as candidate low safety significant, its defense-in-depth is assessed using the following criteria:

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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				

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?

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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 support a system function that is not considered in CDF and LERF, but would be the only means for preserving long-term containment integrity post-core damage (i.e., containment heat removal)?

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In cases where the answer to any of the above questions is "yes," the SSC should be categorized as candidate safety significant. If all of the above questions are answered "no," then low safety significance is confirmed. When complete, if all SSC functions are confirmed as low safety significant, then the SSC remains Condidate Low Safety Significant for the IDP.

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In cases where SSCs are identified as safety significant, the safety significant attributes should be defined. 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.

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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. The overall process used in integrating the various categorization inputs is depicted in Figure 7-1.

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 are 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 or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports.

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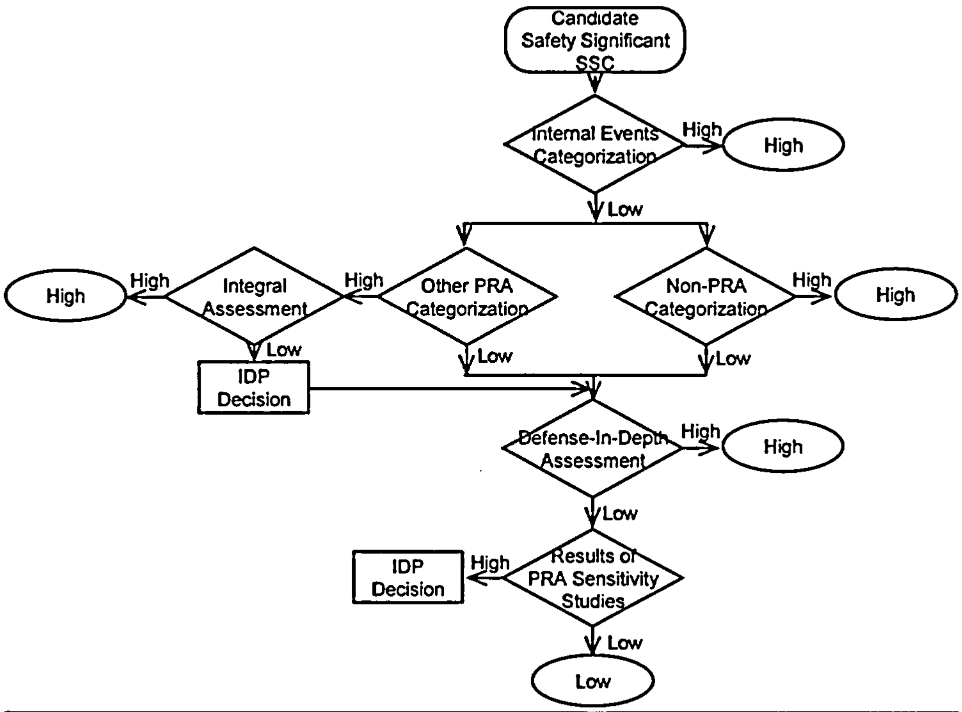
Figure 7-1
Overview of Process for Assigning Overall Safety Significance



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 7-2 provides an example, conceptual layout of the information that summarizes the results and insights that were generated in the categorization process and could be useful for the IDP. This format is for the purposes of identifying the key information that should be communicated to the IDP for use in their decision process. It is expected that additional information will be available at the IDP session that documents the basis for the summary example in the Figure 7-2.

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

- System name
- The function(s) evaluated and the SSCs supporting those functions.
- 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.

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Figure 7-2
EXAMPLE RISK-INFORMED SSC ASSESSMENT WORKSHEET
(FUNCTIONAL BASIS)

System: Function:

Associated Components (or Flowpath):

Function Evaluated for Risk? Yes No

SSCs Modeled (explicitly or implicitly) in Risk Assessments:

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Significance Based on Probabilistic Risk Assessment Tools			
		Potential Risk Significance (High or Low)	Basis for Risk Significance (Include RAW and F-V values where applicable)
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 Individual Sensitivity Studies		
	Change in Risk Significance?	Summary of Findings (Include Delta CDF and LERF or RAW and F-V values where applicable)
Human Error Rates		
Common Cause Failure		
Maintenance Unavailability		
Common Cause Failure		
Others		

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Insights From Cumulative Sensitivity Study for the System:

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Defense-in-Depth Assessment:

Categorization in Other Risk Informed Applications (Maintenance Rule, ISI, etc):

Recommended Categorization for Function:

Safety Significant: Low Safety Significant:

Basis for Categorization:

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8 RISK SENSITIVITY STUDY

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

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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. It is not necessary to address the cumulative impact of SSCs for hazards where screening tools such as SMA were used because if they are included in the screening analysis they are considered high safety significant, thus there would be no change in treatment and no change in performance. For categorizations that rely on PRAs, this sensitivity 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.

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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. In the extreme, individual components could 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.

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The risk sensitivity study should be performed by manipulating the unavailability terms for PRA basic events that correspond to components 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 that could be impacted by the changes in special treatment.

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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.

In cases where the categorization process identifies beyond design basis functions that will be addressed for RISC-1, reducing the unreliability of these safety significant SSCs

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by a similar factor may be called for, depending upon the specific changes in special treatment. The cumulative 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., 0.0025) for a re-evaluation of SSCs risk ranking. This may result in re-categorizing some of the candidate low safety significant SSCs as safety significant SSCs.

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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 risk information relied upon in the categorization process.

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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. This training is for IDP familiarity (i.e., it is not intended to make the IDP PRA "experts").

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, including a list of exempted regulations for low safety significant SSCs,
- The categorization process (e.g., a brief description of Figure 2-1),

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- The risk-informed defense-in-depth philosophy and criteria to maintain this philosophy,
- PRA fundamentals,
- Details of the plant-specific PRA analyses that are relied upon for the preliminary categorization, including

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  - the modeling scope and assumptions,
  - interpretation of risk importance measures, and
  - the role of sensitivity studies and change in risk evaluations

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- The IDP process, including roles and responsibilities.

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¶
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, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS.

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.

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

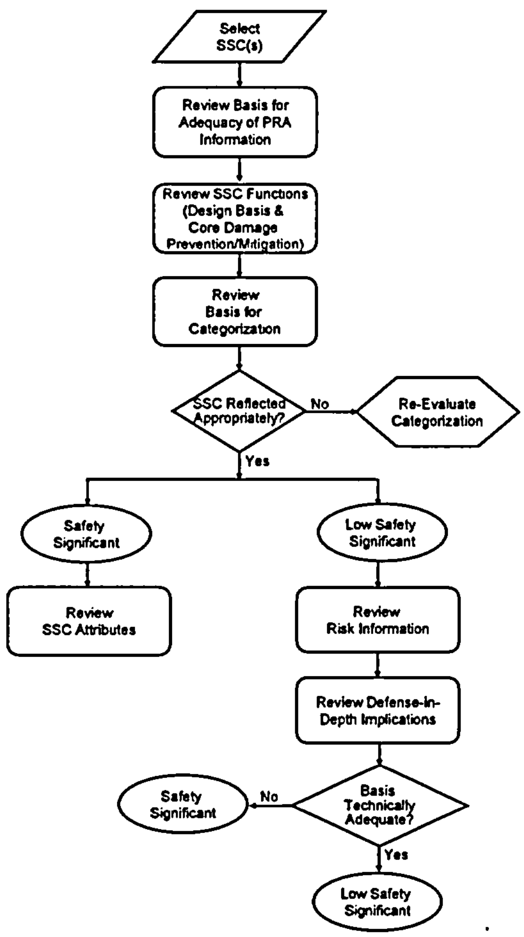
- The designated chairman, panel members, and panel alternates;
- Required training and qualifications for the chairman, members, and alternates;
- Requirements for a quorum, attendance records, agendas, and meeting minutes;
- The decision-making process;
- Documentation and resolution of differing opinions; 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 risk information incorporate those functions. If the IDP determines that the function has not been appropriately reflected, then it is returned to the preliminary categorization process to be re-evaluated based on the insights from the IDP.

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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, then the IDP cannot move that SSC to a low safety significant category. For safety significant 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.

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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 been identified as candidate low safety significant, the IDP should review the results to determine whether these functions/SSCs are not implicitly depended upon for risk-significant functions. The IDP should consider whether:

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- 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 function/SSC will not compromise the reactor coolant pressure boundary or containment integrity.

- Failure of the associated SSC(s) will fail a safety significant 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).). "Function" here is considered to be one of the "high level" general mitigation categories such as "reactivity control", "hi pressure RPV injection from all sources", etc. That is, the IDP reviews the impact of loss of the SSC against the defense-in-depth remaining to perform the function.

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

- 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).

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 may not be adequate if

- The overall redundancy and diversity among the plant's systems and barriers is not sufficient to ensure that no significant increase in risk would occur;

- Reasonable balance is not 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 not 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 an over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design; and

- Potential for common cause failures is not taken into account in the risk analysis categorization.

If any of the above conditions for either the risk information or the defense-in-depth implicdations are true, low safety significance can still be assigned, if the following condition is met:

- Historical data show that these failure modes are unlikely to occur, and

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If any of the above conditions are true, the IDP should

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- Such failure modes can be detected and mitigated in a timely fashion, or
- Condition monitoring – leading indicators

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Deleted: 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

Because the only requirements that are relaxed for low safety significant SSCs are those related to treatment, existing safety margins for SSCs arising from the design technical and functional requirements would remain. It is also required that there be reasonable confidence that any potential increases in CDF and LERF be small from assumed changes in reliability resulting from the treatment changes permitted by 50.69. As a result, individual SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results. Therefore, it can be concluded that the sufficient safety margins are preserved. Consequently, no specific assessment of safety margin is required by the IDP.

Deleted: 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.

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Review of LSS SSCs

The functions/SSCs initially categorized as LSS may 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.

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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 supports 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.

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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, RISC categorizing all SSCs within a system or subsystem according to whether the system or subsystem as a whole performs a risk significant function (Section 10.1). 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.

The two options can be explained in more detail as:

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1) Assignment of all SSCs in the flow path represented by the function to the safety significance classification of that function. While this is a conservative assignment, it may best suit the cost-benefit assessment for 50.69 for a particular system. That is, the effort in going to the next step may not be commensurate with the benefits to be derived.

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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 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:

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- The criterion for assignment of low safety significance for an SSC in a 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

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significance determination for an SSC in a safety significant functional flow path would include, but is not limited to:

- 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 that is not a significant flow diversion path, SSCs downstream of the first 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.).

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IDP Review of RISC 3 and RISC-4 Components

For SSCs that retain the categorization of the function that they support, no IDP review should be required; there should be no differences from the assessments considered in the initial IDP. For SSCs that are re-categorized to a lower classification (e.g., components in a safety significant function that are determined to be low safety significant based on the above considerations), the new categorization and its basis should be presented to another session of the IDP. In this follow-up session, the IDP would be expected to review the basis for the re-categorization and to assess the impact of this re-categorization on the risk importance and defense in depth implications using the same criteria as in the original IDP session for candidate low safety significant SSCs.

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11 PROGRAM DOCUMENTATION AND CHANGE CONTROL

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10 CFR 50.69(f) includes requirements for program documentation, change control and records. In general, the implementation of 10 CFR 50.69 can be divided into two phases: 1) the initial implementation that includes the categorization of SSCs and the application of treatment based on that categorization; and 2) the control of changes to the plant that may impact those SSCs or their categorization basis following the initial implementation. This section provides guidance on meeting the requirements of 10 CFR 50.69(f) for these two phases.

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11.1 Initial Implementation

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The rule requires the licensee or applicant to document the basis for categorization of any SSCs subjected to the categorization process. The heart of this documentation is the procedure used to conduct the categorization process, and a concise summary of the results of the process. For RISC-1 and RISC-2 SSCs, the documentation should include information on any applicable safety-significant beyond design basis functions that were identified. This information is important to the control of any subsequent changes affecting these SSCs following initial implementation. For RISC-3 and RISC-4 SSCs this information should include the basis for concluding that the SSC is low safety significant.

For the purposes of this guidance, initial implementation refers to the first application of the 50.69 rule to a particular system. This may be at the time the first system(s) are categorized under 50.69 or it may be at later time if the licensee chooses a phased approach to categorization wherein only a few systems are categorized each year, for several years.

The rule requires the licensee or applicant to update the FSAR in accordance with 10 CFR 50.71(e) to reflect which systems have been categorized. Following NRC approval to implement 10 CFR 50.69, any changes to the FSAR that reflect alternative treatment of categorized systems should be captured in the licensee's FSAR update process. NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, provides ample guidance on implementing the update process. Any changes to the FSAR associated with initial implementation need not include a supporting review or evaluation under 10 CFR 50.59.

Initial implementation may entail changes to the licensee's quality assurance plan to reflect alternative treatment for categorized systems. Any changes to the quality assurance plan associated with initial implementation need not include a supporting review under 10 CFR 50.54(a). In addition, any regulatory commitments associated with the special treatment requirements in 10 CFR 50.69(b)(1) for SSCs categorized as RISC-3 are no longer applicable to these SSCs and may be dropped at the licensee's discretion.

The waiver of supporting reviews under 10 CFR 50.59 and 10 CFR 50.54(a) is only applicable to the initial implementation of 10 CFR 50.69, i.e., for changes in treatment to

SSCs based on the results of the categorization process. Any other changes to these SSCs are subject to the applicable change control requirements.

11.2 Following Initial Implementation

Subsequent to initial implementation, any changes to alternative treatment for categorized SSCs are subject to applicable change control requirements, e.g., 10 CFR 50.59 and 10 CFR 50.54(a), and must continue to meet the alternative treatment requirements in 10 CFR 50.69.

Changes to categorized SSCs not associated with treatment continue to be governed by the same applicable change control requirements. For RISC-1 and RISC-2 SSCs that have safety-significant beyond design bases functions, the licensee must also maintain reasonable assurance that these functions will be satisfied following the change.

The periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes.

For example, if new information results in a change in categorization of an SSC from RISC-3 to RISC-1, the licensee must reestablish the level of assurance consistent with its safety-significant treatment program that meets the applicable special treatment requirements.

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12 PERIODIC REVIEW

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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 or the defense in depth implications considerations in previous categorizations have been changed as a result of these later categorization activities. If such changes are found, they should be presented to the IDP for consideration in their deliberations on the categorization of the latest system.

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 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:

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- 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 review of the importance measures used for screening in the categorization process[4].

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

[4] 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

In most cases, the categorization would be expected to be unaffected by changes in the plant-specific PRA. However, in some instances, an updated PRA could result in new RAW and F-V importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. In these cases, the assessment of whether a change in categorization is appropriate should be based on the absolute value of the importance measures. The absolute importance is the product of the base CDF/LERF and the importance measure ([RAW-1] or Fussell-Vesely). This is done in order to not inadvertently assess an SSCs as safety significant when its relative importance (FV and RAW) has gone up, but only due to a decrease in overall CDF & LERF. In cases where the importance measures are different between a prior categorization and an updated result, the categorization reassessments of SSCs that have been previously categorized should be based on the following table:

Table 12-1
IMPACT OF PRA UPDATES ON CATEGORIZATION

Prior Categorization	Updated CDF/LERF	Updated Significance Based on Importance	Updated Absolute Importance	Updated 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

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When a change to the categorization of an SSC is suggested either by a change in plant design or operation that would prevent a safety-significant function from being satisfied or by a change in the PRA model as determined from the absolute importance measures, they should be presented to the IDP for concurrence. In these cases, the IDP would assess the basis for the re-categorization by:

- Review of the primary technical bases for the initial categorization, including the system function(s), the risk importance and the basis for their original categorization,
- Review of the technical basis for the change (in palnt deisgn and operation of PRA model) that has resulted in a suggested change to the SSC categorization including the appropriateness of the manner in which the SSC has been reflected as a result of the change, and
- Review of the new risk importance and defense in depth implications.

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The IDP has the final decision regarding the suggested re-categorization.

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13 REFERENCES

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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*

Deleted: ASME Code Case OMN-3, *Requirements for Safety Significance Categorization of Components using Risk-Insights for Inservice Testing of LWR Power Plants*

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Deleted: ASME Code Case N-660*Alternative Repair/Replacement Requirements for Structures, Systems and Components Classified in accordance with Risk-Informed Processes*

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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.

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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

Plant-specific Risk Information – Plant-specific evaluations of beyond design basis capability used in the categorization process including PRAs, FIVE, seismic margins assessments, shutdown safety assessments, etc.

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.***. *Licensee's name and unit number,* takes exception to NEI 00-04 and Reg. Guide1.*** 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*

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.***, 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.

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Provide schedule for implementing SSC categorization¶
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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.¶
¶

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.*** 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 risk information (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.*
- *Site specific seismic hazard curve.*

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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 the life of the facility. These records will be maintained consistent with the *Subject Plant's* configuration control and document management procedure(s) ***X. *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.

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REFERENCES

1. Reg. Guide 1.***, "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.*** 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