NEI 00-04 (Rev 0)

10 CFR 50.69 SSC Categorization Guideline

NUCLEAR ENERGY INSTITUTE

July 2005

SUITE 4001776 I STREET, NW WASHINGTON, DC 20006-3708 202.739.8000

ACKNOWLEDGMENTS

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

NOTICE

TABLE OF CONTENTS

1 INTRODUCTION

This document provides detailed guidance on categorizing structures, systems and components for licensees that choose to adopt 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.* A licensee wishing to implement 10 CFR50.69 makes a submittal to the Director of Nuclear Reactor Regulation, NRC for review and approval. Licensees that commit to implementing 10 CFR 50.69 in accordance with this guideline should expect normal NRC review. Licensees proposing other methods should expect more involved NRC reviews.

The final rule package for 10 CFR 50.69 (including rule language, statements of consideration, regulatory analysis, and other related information) was published in the *Federal Register*, Volume 69, No. 224, November 22, 2004, page 68008. Appendix A of this guidance document provides the rule language for 10 CFR 50.69. Paragraph (a) of the rule provides definitions of terms. Paragraph (b)(1) lists the existing regulations whose scope of applicability is changed through implementation of §50.69. Paragraph (b)(2) describes the contents of the licensee's application. Paragraph (c) describes the categorization process for plant structures systems and components (SSCs). Paragraph (d) provides alternative requirements for the SSCs exempted from the scope of regulations listed in paragraph (b)(1). Paragraph (e) discusses feedback and process adjustment. Paragraph (f) discusses program documentation and change control; and paragraph (g) discusses reporting requirements.

This guidance document is intended to address §50.69 paragraphs (b)(2), (c), (d)(1), (d)(2), (e), and (f). Where appropriate, specific references are provided to the rule paragraphs corresponding to the guidance elements.

This guidance is based on the principles of NRC Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [Ref. 3], 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 10 CFR 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 steps associated with the implementation of 10 CFR 50.69: the categorization of SSCs; and the application of NRC special treatment requirements[1]

[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

consistent with the safety significance of the equipment categorized in the first step. This guidance deals with the categorization of SSCs per 10 CFR 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 10 CFR 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 guidance document 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.

Changes to the categorization process 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 30 years ago, absent data from actual plant operation. It is based on the principle 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 are 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.

---

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.

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 RG 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 SSCs that are governed by NRC special treatment requirements. The latter initiative led to the promulgation of a final rule, 10 CFR 50.69, on November 22, 2004 (see Appendix A for rule language).

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

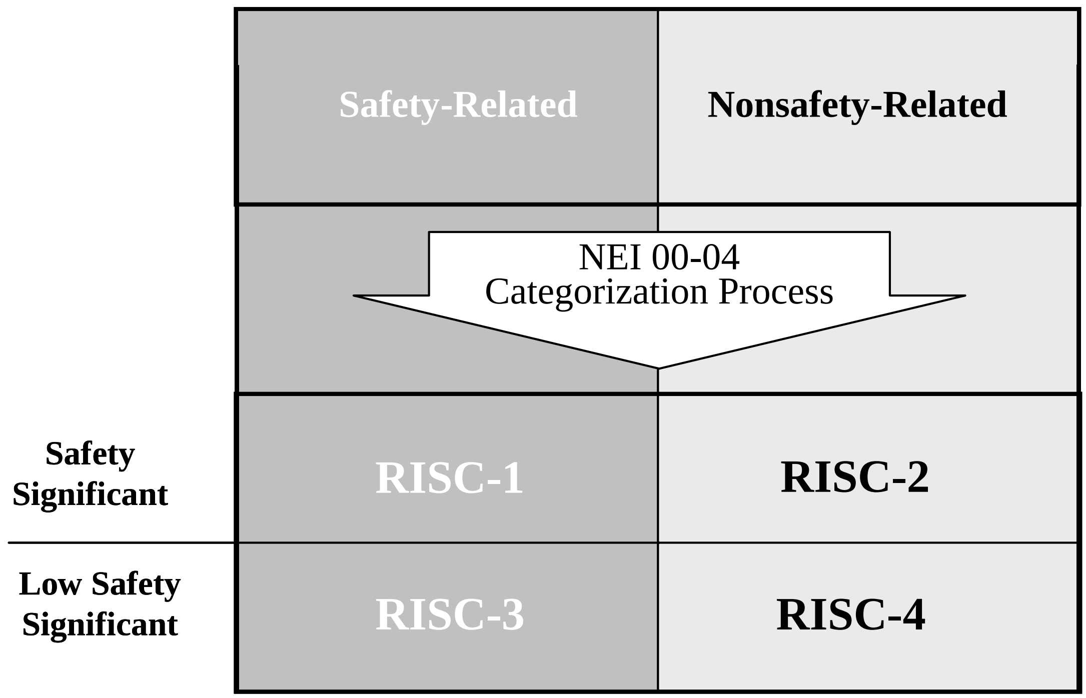
The objective of this regulatory initiative is to adjust the scope of equipment subject to special regulatory treatment (controls) to better focus licensee and NRC attention and resources on equipment that has safety significance. This guideline addresses the use of risk insights to define the scope of equipment that should be subject to NRC special treatment provisions as defined in 10 CFR 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 “non-safety-related,” and is not subject to special treatment requirements. There is a set of non-safety-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.

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

The 10 CFR 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 10 CFR 50.69 categorization process will identify some safety-related SSCs as being of low or no safety-significance (LSS) and these will be categorized as RISC-3 SSCs, while other safety-related SSCs will be identified as safety-significant, and be categorized as RISC-1. Likewise, some non-safety-related SSCs will be categorized as safety-significant (RISC-2) and others will remain of low or no safety significance, and be categorized as RISC-4 SSCs. For the purposes of implementing 10 CFR 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 non-safety-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 10 CFR 50.69 SSC categorization process remain as safety-related, non-safety-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.
- If PRA information is available, the categorization process should employ a blended approach considering both quantitative PRA information and qualitative information.
- The RG 1.174 principles of the risk-informed approach to regulations should be maintained.
- A safety-related SSC will be categorized as RISC-1 unless a basis can be developed for categorizing it as RISC-3.
- Attribute(s) that make a SSC safety-significant, and the basis for categorization as LSS, should be documented.

1.4 VOLUNTARY AND SELECTIVE IMPLEMENTATION

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 10 CFR 50.69 categorization process would provide a benefit. As a result, a licensee can determine the appropriate set of equipment to categorize under 10 CFR 50.69, and schedule the implementation over a period of time. Implementation should be conducted on entire systems/structures, not selected components within a system. The primary reason that 10 CFR 50.69 requires the categorization to be performed for entire systems and structures is to ensure that all functions (which are primarily a system-level attribute) for a given SSC within a given system or structure are appropriately considered for each SSC in determining its safety significance.

1.5 CATEGORIZATION PROCESS SUMMARY

The NEI 00-04 categorization process embodies the principles of risk-informed regulation described in RG 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). SSCs identified as HSS by any of the following may not be re-categorized by the IDP:

- An SSC identified as HSS by the risk characterization portion of the process (which addresses internal events, external events, shutdown, and integrated importance),

- An SSC identified as HSS by the internal events PRA assessment,

- An SSC identified as HSS by a non-PRA method to address external events, fire, seismic, or shutdown,

- An SSC identified as HSS by the defense in depth assessment.

SSCs not meeting any of the above, but identified as HSS through a seismic PRA, external events PRA, fire PRA, shutdown PRA, or through the sensitivity studies in Section 5, may be presented to the IDP for categorization as LSS, if this determination is supported by the integrated assessment process and other elements of the categorization process.

The IDP function is to review the assessment and ensure that the system functions and operating experience have been appropriately considered in the risk analyses.

SSCs that are safety related and considered to be 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 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.

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 believes 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 LSS.

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 may mask the results of individual risk contributors.

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 PRA with appropriate technical capability 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 required for 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 LSS is found to exceed the safety significance thresholds in a sensitivity study, this information is provided to the IDP for consideration, along with an explanation of the results of the sensitivity study.

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 reqired for 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 the FIVE analysis 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 ensures that the SSCs that were required to maintain low fire risk are retained as safety-significant.

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 required for 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.

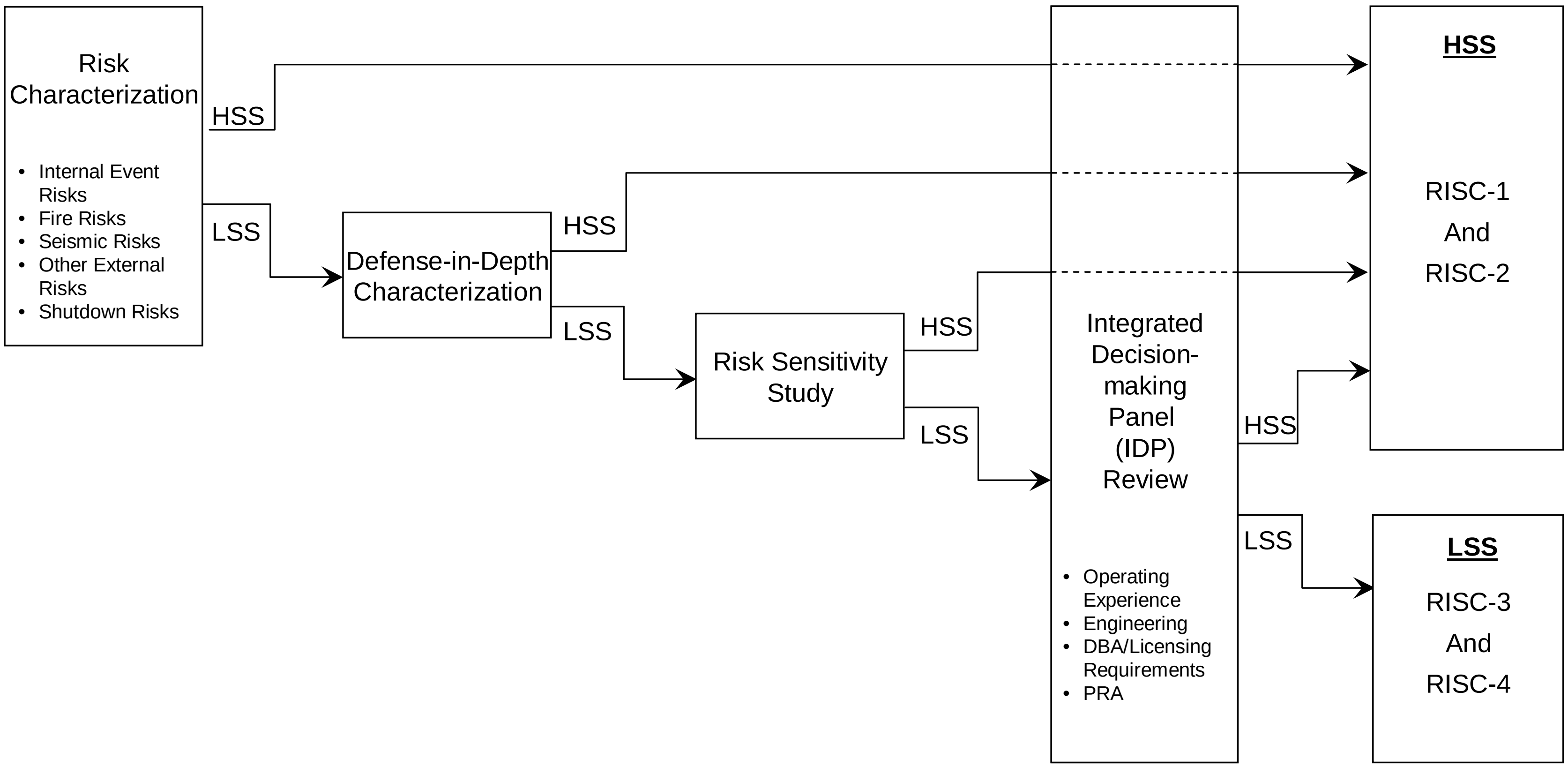
Note, if a seismic PRA is used, SSCs may have been screened out of the PRA due to inherent seismic robustness. For such screened SSCs, regardless of their categorization outcome, it is important that the inherent seismic robustness that allows them to be screened out of the seismic PRA should be retained. This is necessary to maintain the validity of the categorization process.

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 the SMA analysis 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 (High Winds, External Floods, etc)

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 required for 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.

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



In the event a 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 required for 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 shutdown 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.

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

Integrated Importance Assessment

Each risk contributor is initially evaluated separately in order to avoid reliance on a combined result that may mask the results of individual risk contributors. The potential masking is 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 LSS (i.e., 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 ensure 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 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 modeled 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 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 because the categorization processes for hazards that do not have a PRA is done in a manner that ensures the risk sensitive SSCs are categorized as safety-significant. For example, when 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 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 ensure 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/function that is currently safety-related, then the defense in depth assessment must also have shown that the SSC/function is not safety-significant. Finally, the risk sensitivity study verifies that the combined impact of a postulated simultaneous degradation in reliability of all LSS SSCs would not result in a significant increase in CDF or LERF.

If an SSC/function 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 overrule 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 evaluated 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 this document, ensures 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 organizations (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

Rule Reference:

10 CFR 50.96(b)(2) states the following:

> *(2) A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:*
>
> *(i) A description of the process for categorization of RISC-1, RISC-2, RISC-3 and RISC-4 SSCs.*
>
> *(ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant- specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.*
>
> *(iii) Results of the PRA review process conducted to meet § 50.69 (c)(1)(i).*
>
> *(iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69 (c)(1)(iv). The evaluations shall include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).*

NRC Regulatory Guide 1.201 provides that the categorization process described in this document (with any noted exceptions or clarifications) is acceptable for implementation of 10 CFR 50.69. The description required by §50.69(b)(2)(i) may be limited to a statement that the process of NEI-00-04 was followed. Any exceptions or alternative methods to portions of the NEI-00-04 process should be noted and justified.

The remaining requirements of §50.69(b) above are addressed in Sections 2, 3, and 8 of this document, as follows:

- Sections 3.2 and 3.3 provide expectations and general guidance for assessment of PRA scope and technical capability to address §50.69(b)(ii) and (iii).
- Section 8 provides specific guidance corresponding to §50.69(b)(iv).

10 CFR50.69(c) states the following:

(1) SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines whether an SSC performs one or more safety- significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth.

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § 50.69(b)(1) and § 50.69(d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

(2) The SSCs must be categorized by an Integrated Decision- Making Panel (IDP) staffed with expert, plant- knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

The requirements of §50.69(c) above are met by the overall process depicted in Figure 2-1, and described in Sections 2 through 10 of this document, as follows:

- Sections 3.2 and 5.1 provide specific guidance corresponding to §50.69(c)(1)(i).
- Sections 3, 4, 5, and 7 provide specific guidance corresponding to §50.69(c)(1)(ii).

- Section 6 provides specific guidance corresponding to §50.69(c)(1)(iii).

- Section 8, provides specific guidance corresponding to §50.69(c)(1)(iv).

- Section 2 provides specific guidance corresponding to §50.69(c)(1)(v).

- Sections 9 and 10 provide specific guidance corresponding to §50.69(c)(2).

The categorization 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.

Select Scope of Plant Systems for §50.69 Categorization

Rule Reference: 10 CFR 50.69(c)(1)(v)

10 CFR 50.69 does not require a specific scope of implementation. Thus, the licensee may select the systems for which §50.69 would be implemented, and may conduct the implementation in a phased manner. However; §50.69(c)(1)(v) provides that implementation should be conducted on entire systems/structures, not selected components within a system. The primary reason that 10 CFR 50.69 requires the categorization to be performed for entire systems and structures is to ensure that all functions (which are primarily a system-level attribute) for a given SSC within a given system or structure are appropriately considered for each SSC in determining its safety significance. System boundary definitions are important in addressing safety functions, especially those functions that may involve multiple systems. System boundary definitions should be developed by the licensee, consistent with those used in the PRA supporting categorization.

The categorization 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 sections 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 ensure 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, Maintenance Rule assessments and PRA analyses. The mapping of components is performed to allow the correlation of PRA importance measures to system functions. More detail on this step is provided in Section 4.

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 on this step is provided 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 on this step is provided in Section 6.

Preliminary Engineering Categorization of Functions

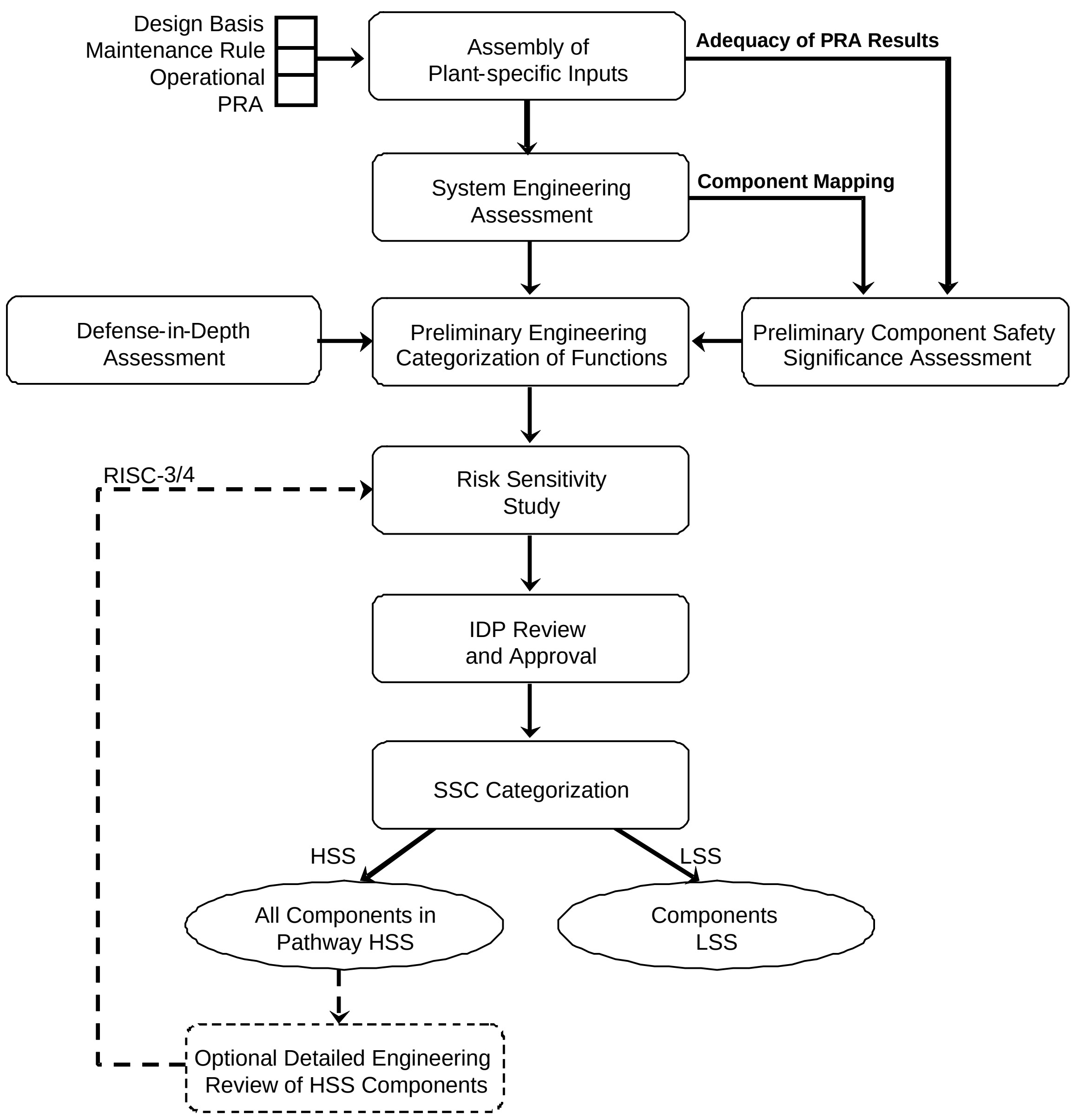
This step involves integrating the results of the two previous tasks to provide a preliminary categorization of the safety significance of system functions. This includes consideration of both the risk insights and defense-in-depth assessments. More detail on this step is provided in Section 7.

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 on this step is provided in Section 8.

Figure 2-1

RISK-INFORMED CATEGORIZATION PROCESS



IDP Review and Approval

The IDP is a multi-disciplined team that reviews the information developed by the categorization team. The 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 assessment to finalize the categorization of functions. More detail on this step is provided in Section 9.

SSC Categorization

When the IDP approves the categorization of system functions, then the initial coarse mapping of components to system functions may be used to define the safety significance of each SSC. 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 LSS because a failure of these SSCs would not inhibit a safety significant function. In the event this more detailed review identifies any HSS SSCs that can be categorized as LSS, the results of that re-categorization are re-evaluated in the risk sensitivity study and provided to the IDP for final review and approval. More detail on this step is provided 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, and
- 10 CFR 50.65 Information

3.2 Use of Risk Information

Rule Reference

10 CFR 50.69(b)(2)(ii) and (iii) state the following:

> *(2) A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:*
>
> *(ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant- specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.*

(iii) Results of the PRA review process conducted to meet § 50.69 (c)(1)(i).

§50.69(c)(1)(i) states:

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

§50.69(b)(2)(ii) and (iii) require that the application contain a description of the measures taken to assure sufficient quality and level of detail for the risk analyses used to support implementation. This includes the internal events at power PRA required by §50.69(c)(1)(i), as well as the risk analyses used to address external events and shutdown conditions.

Section 3.2 of this document discusses the use of risk information and the general quality measures that should be applied to the risk analyses supporting the 50.69 application.

Section 3.3 of this document discusses the characterization of adequacy of both the internal events at power PRA, as well as other risk analyses necessary to implement 50.69, and the specific information that should be presented to the IDP in this regard.

To satisfy the requirements of §50.69(b)(2)(ii) and (iii), the application should contain a description of the above measures, specifically the items listed in sections 3.3.1 and 3.3.2.

An essential element of the SSC categorization process is a plant-specific full power internal events PRA, which 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 to the IDP, 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 RG 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.

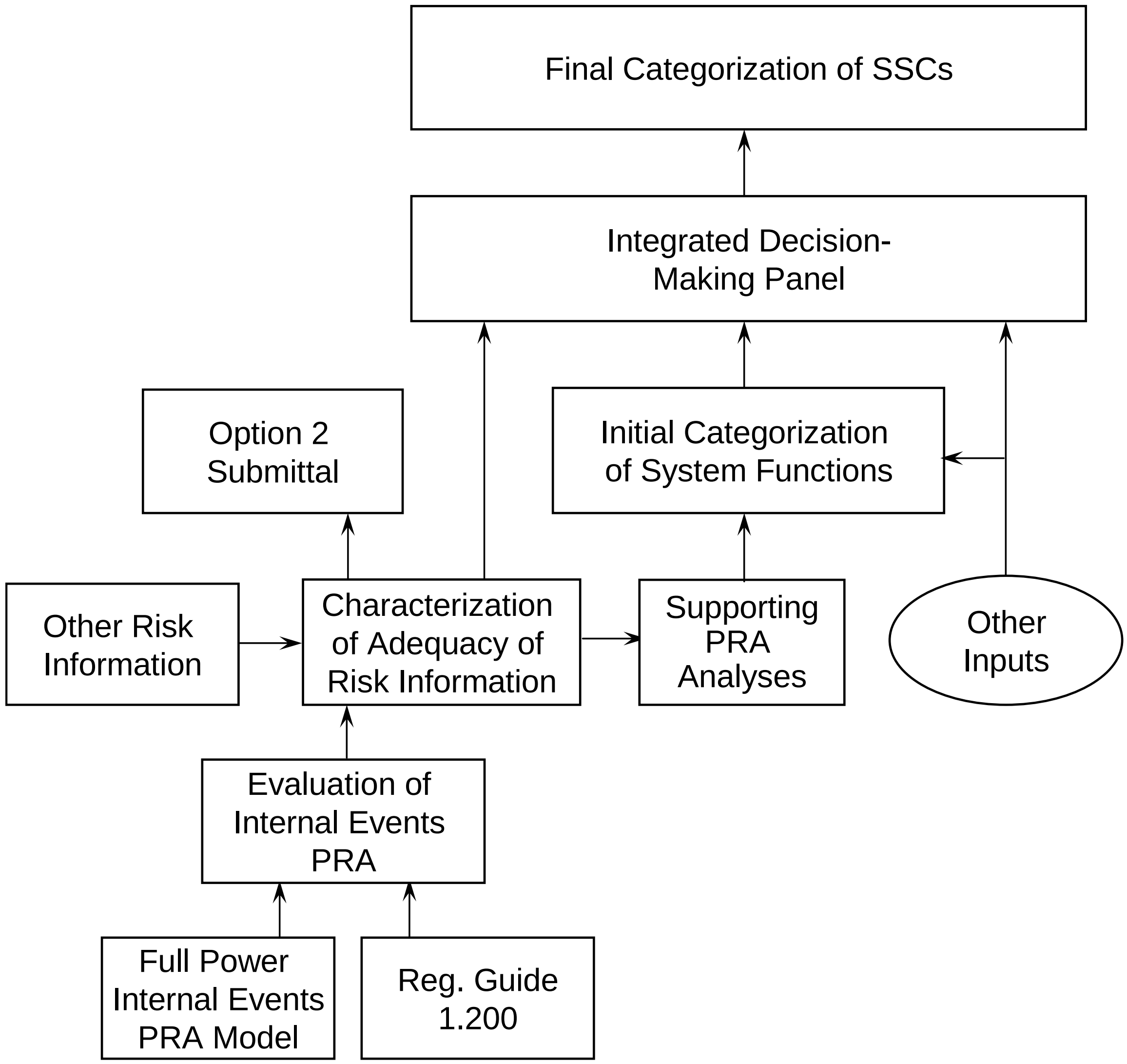
3.3 Characterization of the Adequacy of Risk Information

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 full power internal events PRA.

The primary basis for evaluating the technical adequacy of PRA studies relies upon Regulatory Guide 1.200, ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.'' [Ref 11] 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'' Ref 12). 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 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.

Figure 3-1

PROCESS FOR ASSURING PRA ADEQUACY
FOR 10 CFR 50.69 CATEGORIZATION



In addition, it may be useful for the licensee to consider the guidance provided by the NRC staff in a letter to NEI dated April 2, 2002, [Ref. 16], ADAMS accession number ML020930632. This letter provides draft staff review guidance that was developed as a result of its review of NEI 00–02 for intended use for 10 CFR 50.69 applications.

Peer review findings are a significant part of justifying 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:

3.3.1 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 in the NEI 00-02 process, or lower than ASME Capability Category II in the Reg Guide 1.200 process.
- A high level summary of the results of the self assessment process discussed in Reg Guide 1.200
- 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. 16]. This NRC letter discusses PRA technical attributes that are important to the 10 CFR 50.69 application, as well as those that are of lesser importance. The specific PRA attributes are identified in the context of NEI-00-02, but may be applied to corresponding elements of the ASME standard and Reg Guide 1.200. This information will be useful in addressing Reg. Guide 1.200, which specifies that PRA technical attributes pertinent to a given application will be identified.

3.3.2 Other Risk Information (including other PRAs and screening methods)

- A basis for why the other risk information adequately reflects 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 IDP should be aware of the limitations of the PRA to support the categorization of the functions/SSCs of the system being considered. The process to be used to justify the adequacy of the risk information is also summarized in the submittal to the NRC.

4 SYSTEM ENGINEERING ASSESSMENT

Rule Reference:

The guidance in this section (in combination with other sections) addresses §50.69(c)(1)(ii). See discussion in Section 2 for further information.

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 within the defined boundaries 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.

An SSC shall be categorized as safety significant if it is safety-significant for the particular system being considered. However, there may be circumstances where the categorization of a candidate low safety-significant SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. In this case, the SSC will remain uncategorized until the interfacing system is considered. For example, cooling water system piping on a ventilation system cooler is designated as part of the ventilation system. The impact of failure of the SSC on the ventilation system can be considered, but the impact of failure of the SSC on the cooling water system cannot be fully assessed until that system is considered as part of a future categorization process. Therefore, the SSC will remain uncategorized and continue to receive its current level of treatment requirements.

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 as appropriate. (Some maintenance rule functions may not correlate directly with the 50.69 system functions). While beyond design basis functions may be included in the maintenance rule functions, a review of the PRA should be conducted to ensure 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 LSS components associated with a given function. Additional functions may be identified (e.g., fill and drain) to group and consider potentially low safety-significant components that may have been initially associated with a safety-significant function but which do not support the critical attributes of that safety-significant function.

The classification of SSCs having only a pressure retaining function (also referred to as passive components), or the passive function of active components, should be performed using the ASME Code Case N-660, "*Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities*" (Ref. 17), or subsequent versions approved by ASME, in lieu of this guidance.

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 function are identified and documented. Several options may be included in this implementation element:

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

2) If passive components have been categorized according to guidance for risk-informed inservice inspection (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, is being implemented at the same time. [2]

   In these cases, for each of the system functions from the previous step, the ISI

[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 50.69 submittal.

segments associated with that function must be defined. That is, the pathway 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 a conservative approach because not every component associated with 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 pathway, but the components associated with these functions can be readily identified from system drawings once the system boundaries are identified.

The assignment of SSCs to 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. In subsequent steps, the categorization of all system functions will be performed and will be presented to the IDP for review. The categorization assigned to each of the system functions will initially be applied to the SSCs associated with that function. The detailed categorization process of Section 10.2 may then be applied to further refine the categorization based on other considerations that may make the safety significance of an SSC lower than that of the initially associated function.

5 COMPONENT SAFETY SIGNIFICANCE ASSESSMENT

Rule Reference:

The guidance in this section (in combination with other sections) addresses §50.69(c)(1)(ii). Section 5.1 additionally addresses §50.69(c)(1)(i). See Section 2 for further information.

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.

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, including containment functions, then the risk screening process is terminated and the system functions are categorized as candidate LSS. However, the system/structure must still be assessed for defense in depth considerations and presented to the IDP.

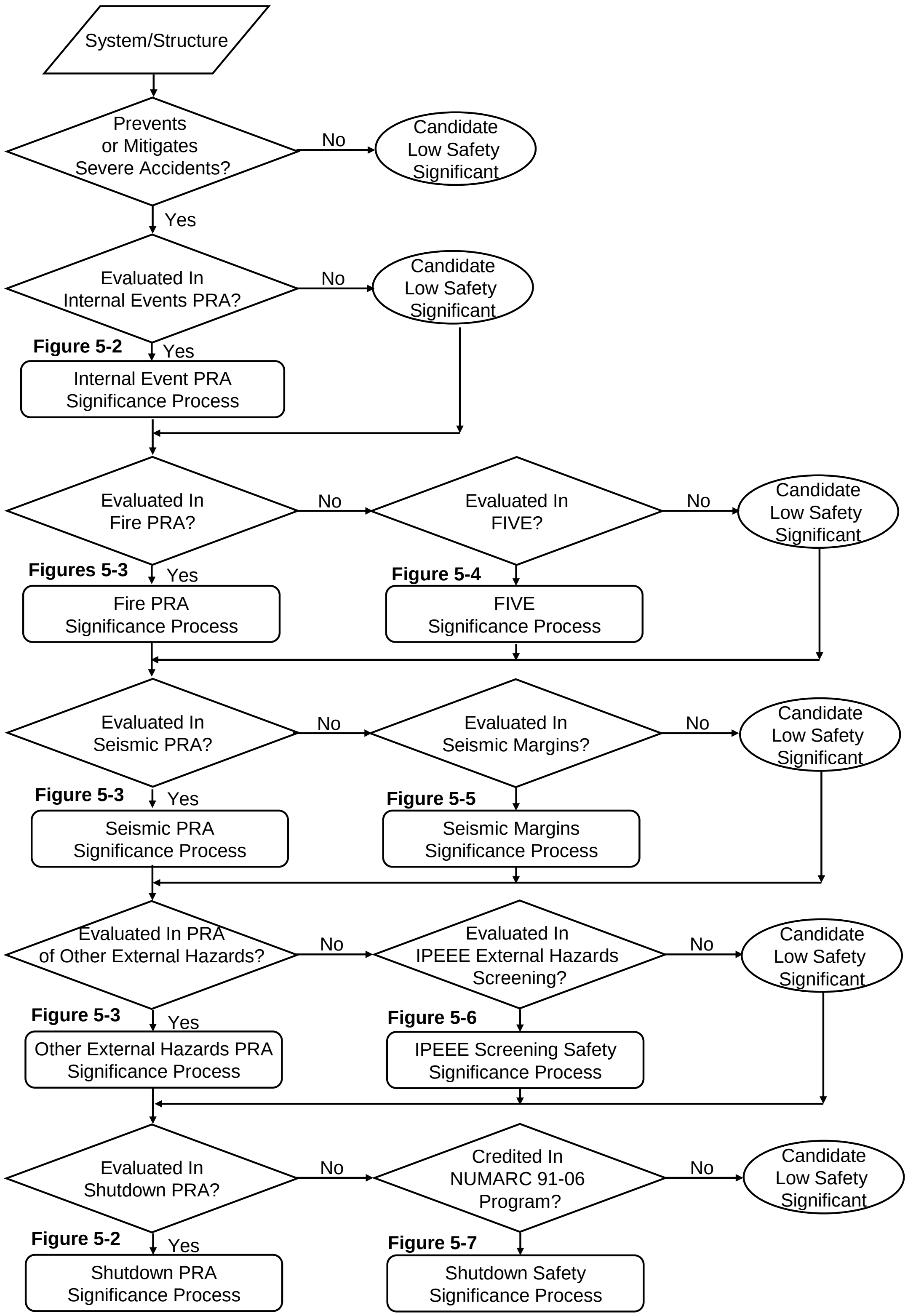
Significance from Internal Events

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 its failure contribute to an 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

Figure 5-1
USE OF RISK ANALYSES FOR SSC CATEGORIZATION



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 LSS from the standpoint of internal event risks. The evaluation is continued with fire risk.

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 analysis (SMA) 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 SMA. If the system or structure is included in the SMA, 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 SMA, then the SSC is categorized as candidate LSS from the standpoint of seismic risk.

Significance from Other External Events

If the plant has a PRA that 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 an external hazards PRA or external hazards screening evaluation, then the SSC is categorized as candidate LSS 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 LSS from the standpoint of shutdown risk.

5.1 Internal Events 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 uses two standard PRA importance measures, risk achievement worth (RAW) and Fussell-Vesely (F-V), as screening tools to identify candidate safety-significant SSCs. The criteria chosen for safety significance using these

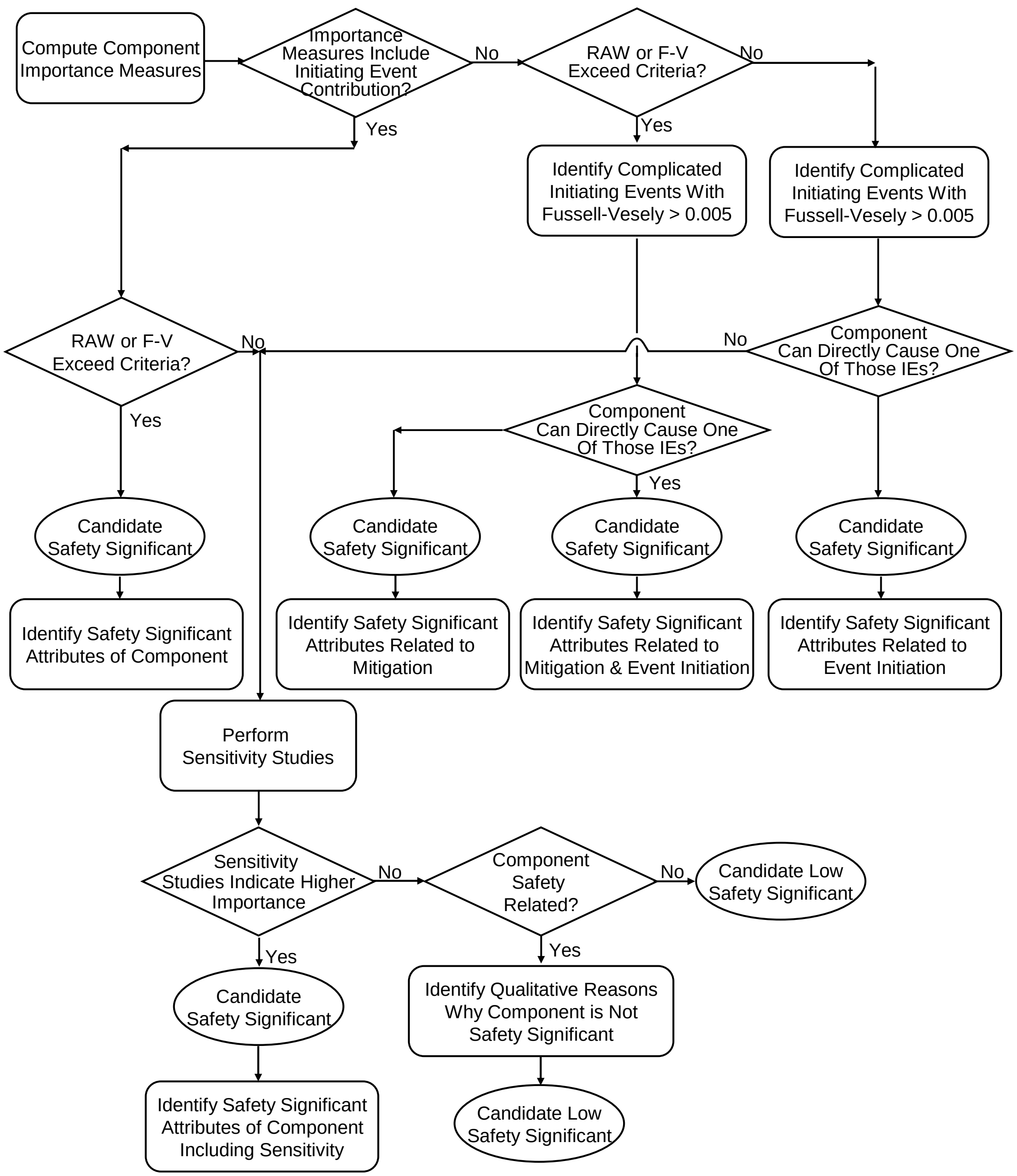
---

[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.

importance measures are based on previously accepted values for similar applications. Risk reduction worth (RRW) is also an acceptable measure in place of F-V

Figure 5-2

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



because the F-V criteria can be readily converted to RRW criteria. The F-V importance of a component is considered to be the sum of the F-V importances for the failure modes of the component relevant to the function being evaluated.

If a component does not have a common cause event to be included in the computation of importances, then an assessment should be made as to whether a common cause event should be added to the model. The RAW importance of a component is considered the maximum of the RAW values computed for basic events involving 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 is considered separately from the basic events that reflect an individual component. A RAW value of 20 was conservatively selected to reflect the fact that the common cause RAW is measuring the failure of two or more trains, including the higher failure likelihood for the second train due to common causes. 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.

An analysis of the impacts of parametric uncertainties on the importance measures used in this categorization process was performed and documented in EPRI TR-1008905, *Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization* [Ref. 17]. The conclusion of this analysis was that the importance measures used in combination with identified set of minimum sensitivity studies adequately address parametric uncertainties.

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 example in Table 5-1 above, 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 F-V 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 significantly contributing failure modes would be used in defining the attributes.

In cases where the internal events CDF 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 scenarios. 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 LSS. 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 F-V 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 LSS 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 F-V risk importance measure, it is recommended that a CDF (or LERF) truncation level of 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. The selected truncation level should support an overall CDF/LERF that has converged, and must be within the capability of the software used. In addition, the truncation level used should be sufficient to identify all functions 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 solution identifies all safety-significant functions. 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

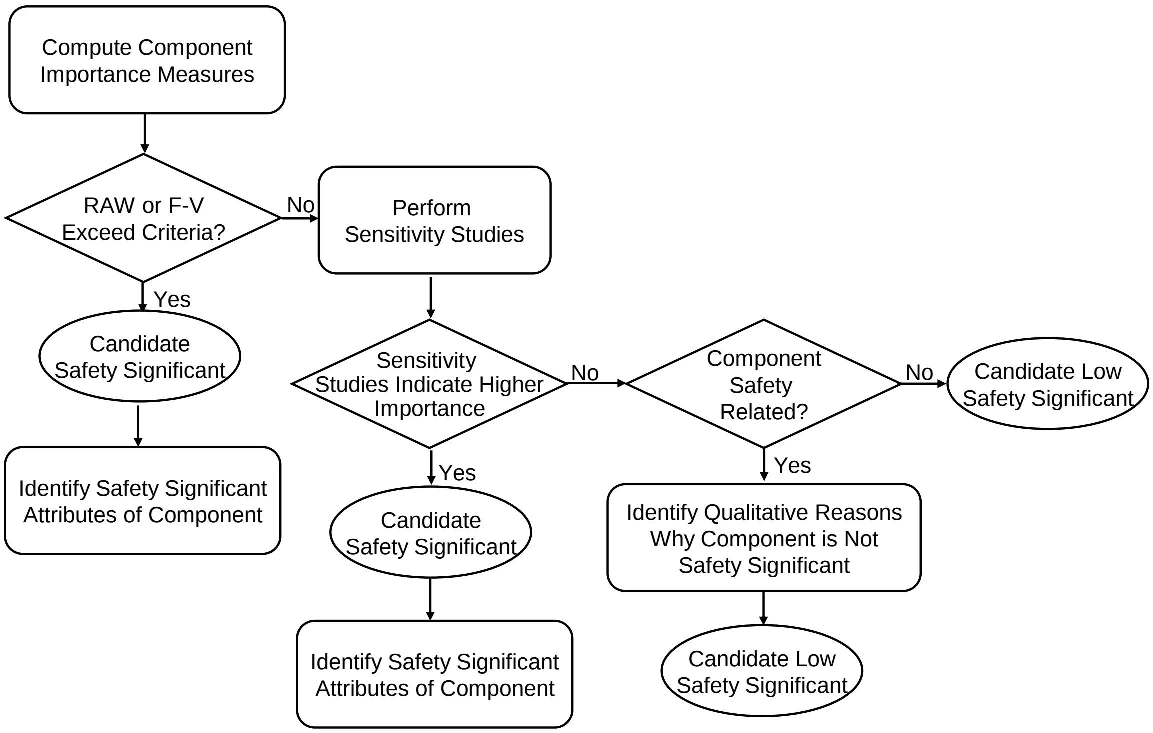
The fire safety significance process takes one of two forms. For plants with a fire PRA, the process is similar to that described for an internal events PRA. This process is shown on Figure 5-3, and is discussed below. Plants that relied upon a FIVE analysis to assess fire risks for the IPEEE should 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 F-V 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, including all screened scenarios, 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 LSS 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., does not 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.

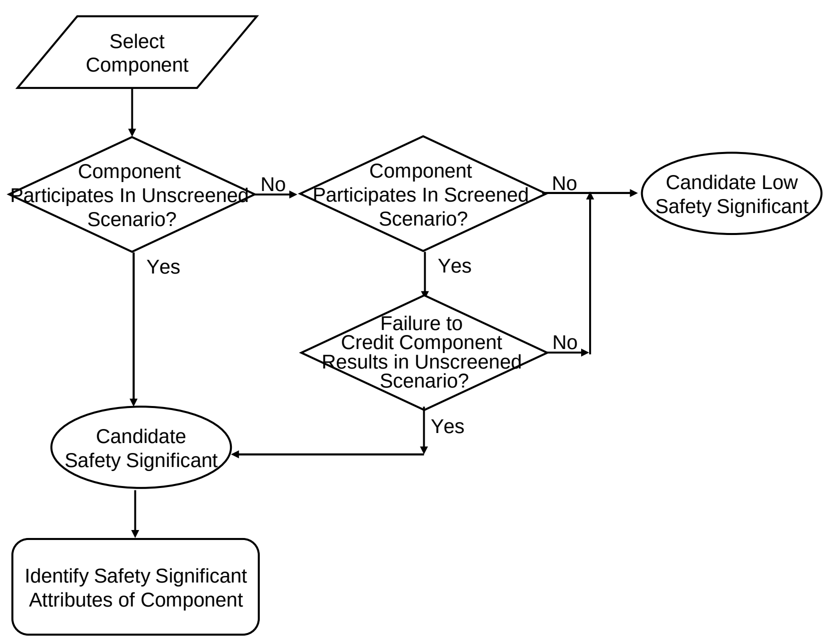
The recommended sensitivity studies for fire PRA are identified in Table 5-3.

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 • No credit for manual suppression • Any applicable sensitivity studies identified in the characterization of PRA adequacy

The FIVE methodology is a screening approach to evaluate 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 LSS 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 that plant components cannot 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 LSS from a seismic perspective.

Note, if a seismic PRA is used, SSCs may have been screened out of the PRA due to inherent seismic robustness. For such screened SSCs, regardless of their categorization outcome, it is important that the inherent seismic robustness that allows them to be screened out of the seismic PRA should be retained. This is necessary to maintain the validity of the categorization process.

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 LSS 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.

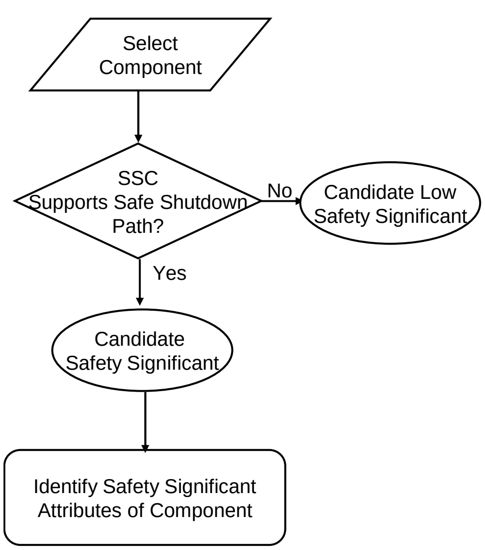
The recommended sensitivity studies for seismic PRA are identified in Table 5-4.

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 LSS 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 this conservatism.

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.

The generalized safety significance process for plants with an external hazard PRA is the same as the process for an internal events PRA. As with 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 LSS 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

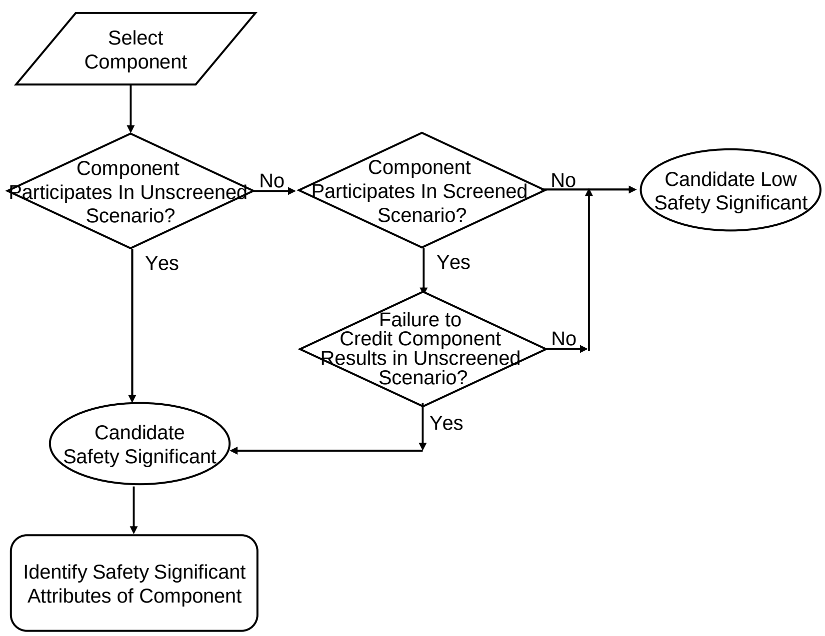
Sensitivity Study
• Increase all human error basic events to their 95^{th} percentile value • Decrease all human error basic events to their 5^{th} percentile value • Increase all component common cause events to their 95^{th} percentile value • Decrease all component common cause events to their 5^{th} percentile value • 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 that yielded that conclusion should be identified. If, following the sensitivity studies, the SSC is still found to be LSS and it is safety-related, the analyst is expected to define why that component is of low risk significance (e.g., does not 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 LSS components. The safety significance process for plants with external hazard screening evaluations is shown in Figure 5-6.

Figure 5-6
OTHER EXTERNAL HAZARDS



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 the component is credited, it is considered safety-significant 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 external hazard screening process does not generate numerical estimates of core damage frequency values. However, the option always exists for the licensee to perform an external events PRA to remove this conservatism.

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 LSS 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.

In this process a component can be identified as safety-significant for shutdown conditions for either of the following reasons:

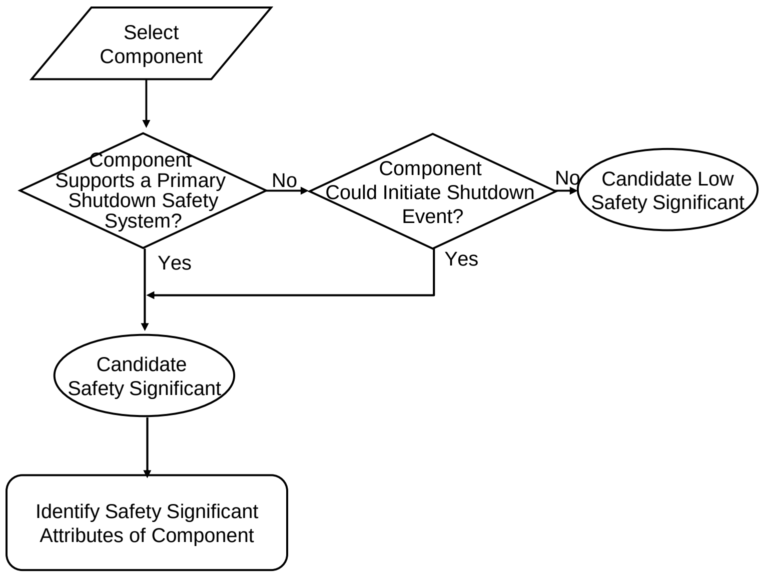
- NUMARC 91-06 specifies that a defense in depth approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and alternative system/train to accomplish the given key safety function. When multiple systems/trains are available to satisfy the key safety function, only SSCs that support the primary and first alternative methods to satisfy the key safety function are considered to be the “primary shutdown safety system” and are thus candidate safety-significant .

.

- Its failure would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.),

Figure 5-7

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



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.

5.6 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 process 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 F-V Importance of Component i over all CDF Contributors
$FV_{i,j}$ = F-V 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 F-V >0.005, RAW > 2.0 for individual basic events, and RAW > 20 for common cause basic events. 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

Rule Reference:

The guidance in this section addresses §50.69(c)(1)(iii). See Section 2 for further information.

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 an SSC as safety-related) and considers the level of defense-in-depth available, based on the success criteria used 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 LSS,

- Identify the design basis events for which the function is required.
- 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. Potential combinations of other systems and trains are depicted across the top row of Figure 6-1. Credit may be taken for systems containing RISC-1, 2, 3, or 4 SSCs (with the exception noted in the bullet below), and realistic success paths may be used.
- For each design basis event, identify the region of Figure 6-1 in which the plant mitigation capability lies **without** credit for the function/SSC that has been proposed as low safety-significant, and without credit for any identical, redundant SSCs within the system that are 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 function/SSC has been confirmed.
- If the result is in the region entitled “Potentially Safety-significant,” then the function/SSC should be classified as safety-significant for the IDP, noting that the basis is core damage defense-in-depth.

When complete, if all SSC functions are confirmed as LSS, 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 LSS 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 redundant systems are required or three or more trains must exist. For BWRs, there are multiple coolant inventory makeup systems that could be used without crediting LPCS (i.e., HPCI, Reactor Core Isolation Cooling (RCIC), main feedwater, condensate, and LPCI with Automatic Depressurization System (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 control rod drive cooling (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 large containment failures (e.g., due to loss of containment heat removal). For each SSC function categorized as candidate LSS, 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 an ISLOCA event?
- Can the SSC provide a significant level of mitigation of an ISLOCA event? [Note that mitigation (up to and including isolation) of ISLOCA is a beyond design basis function. There are a number of SSCs that could be credited with providing varying degrees of mitigation of an ISLOCA. However, only SSCs providing a significant level of mitigation should be candidate HSS. These SSCs would also be treated in the internal events model as LERF mitigators, and thus their significance would be considered in that aspect of the categorization process.]
- 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:
  - Directly connected to containment atmosphere, and
  - > 2" in diameter, and
  - not locked closed or only locally operated?
- Does the SSC support containment isolation for containment penetrations that are:
  - Part of the reactor coolant system pressure boundary, and
  - > 3/8" in diameter, and
  - not locked closed or only locally operated?

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 (e.g., 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 candidate LSS 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

Rule Reference:

The guidance in this section (in combination with other sections) addresses §50.69(c)(1)(ii). See Section 2 for further information.

7.1 Engineering Categorization

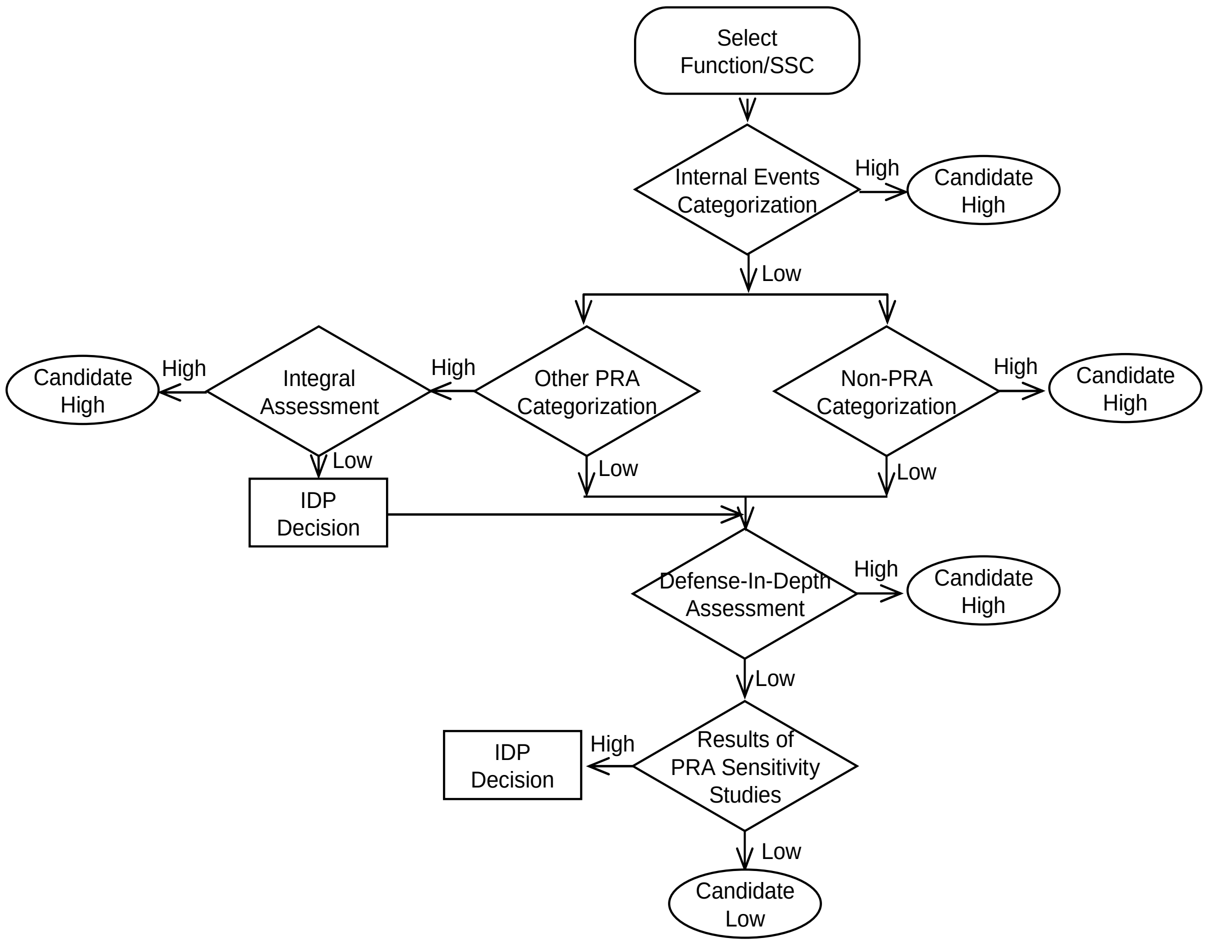
This step involves the assignment of preliminary safety significance to each of the functions identified previously. The safety-significant SSCs from the component safety significance assessment (Section 5) are mapped to the appropriate function for which they had safety significance. If any SSC is safety significant, from either the PRA-based component safety significance assessment (Section 5) or the defense-in-depth assessment (Section 6), then the associated system function is preliminarily safety significant. All other functions/SSCs can be preliminarily assigned low safety significance. All preliminary categorizations assigned as candidate safety significant or low safety significant are then taken to the IDP for final review and approval. 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 that support this system function are assigned a preliminary safety-significant categorization. All other components are assigned a preliminary LSS 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.

For safety-significant functions/SSCs, the critical attributes that make the function/SSC safety-significant need to be identified. Critical attributes should include high level features of the SSCs that contribute to the safety significance of the function, such as provide flow, isolate flow, etc. These "critical" attributes provide information to the treatment activity implementers to assure that correct levels of treatment requirements are applied to monitor or maintain the SSC critical attributes. The identification of important-to-safety attributes may also be used as a means of justification for RISC-2 categorization of non-safety-related SSCs.

Figure 7-1
Overview of Process for Assigning Preliminary 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 significance 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 results of these PRA assessments, as well as the integral assessment should be presented to the IDP.

- 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: ____

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):_______________
____
____

Recommended Categorization for Function:

Safety-significant: ___________ Low Safety-significant: ________

Basis for Categorization:____
____

8 RISK SENSITIVITY STUDY

Rule reference:

The guidance in this section, in combination with the overall process described in sections 2 through 7 of this document, addresses §50.69(c)(1)(iv) and §50.69(b)(2)(iv).

§50.69(c)(1)(iv) states the following

> *Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § 50.69(b)(1) and § 50.69(d)(2) are small.*

Additionally, §50.69(b)(2)(iv), which discusses contents of the application, states the application must include the following:

> *A description of, and basis for acceptability of, the evaluations to be conducted to satisfy §50.69 (c)(1)(iv). The evaluations shall include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).*

The risk sensitivity study is used to confirm that the categorization process results in acceptably small increases to CDF and LERF. This study is the final step in the categorization process; however, the entire risk evaluation process is integral to performance of the sensitivity study. That is, the set of SSCs whose unreliability is adjusted in the final sensitivity study is determined through the overall process. described in Sections 2 through 7. Without the preceding steps in the process, the risk sensitivity study obviously could not be performed. Thus, all the elements of the process are considered to be part of the “evaluations” that are referenced in §50.69(c)(1)(iv).

The overall risk evaluation process addresses both known degradation mechanisms and common cause interactions, and meets the requirements of §50.69(b)(2)(iv). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data, and provide timely insights into the need to account for any important new degradation mechanisms.

With regard to common cause interaction, this is addressed in both the base PRA model, and in the categorization process, as follows:

- Common cause treatment in the base PRA supporting the process must meet the requirements in the ASME Internal Events at Power PRA Standard . For example, criteria for treatment of common cause are delineated in 16 supporting requirements in the ASME PRA standard associated with HLR-SY-B. The expectation is that the PRA will include a robust evaluation of common cause in the base model, equivalent to that specified for Capability Category II of the ASME PRA Standard for the relevant supporting requirements. PRA evaluation of common cause focuses on intra-system common cause failures, because operating experience has shown that common cause failures occur within a system and generally are a function of service conditions or maintenance practices. These factors are recognized as having the potential to affect similar components within systems. Inter-system common cause failures are normally not modeled because other factors, such as design diversity and different service environments, ensure their negligible contribution to overall plant risk.
- The risk evaluation process (see Section 5) requires consideration of common cause risk importance measures, both RAW and F-V. This is used to assure that groups of components with potentially high common cause impacts are maintained in the RISC-1 or 2 categories.
- The defense in depth evaluation conservatively assumes that the proposed RISC-3 SSCs being evaluated, and any redundant identical RISC-3 SSCs in the system, do not perform their function, and assures that key safety functions are still maintained by redundant SSCs.
- The integrated risk sensitivity study conservatively increases the failure rate of all RISC-3 SSCs simultaneously to assure that potential increases in delta CDF and delta LERF due to changes in treatment are small.
- The corrective action requirement in 10 CFR 50.69(d)(2)(ii) specifically addresses conditions adverse to quality (e.g., common cause failures), and requires that the cause of the condition be determined and corrective action taken to preclude repetition.
- Performance monitoring will ensure that potential increases in failure rates will be detected and addressed before reaching the rate assumed in the sensitivity study.

8.1 Sensitivity Study Process

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 LSS SSCs by a factor of 3 to 5 could provide an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of all LSS SSCs. Such 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 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 function. The basic events for both random and common cause failure events should be increased for failure modes of the component relevant to the function being considered.

In identifying the specific factor to be used in the risk sensitivity study, two considerations should be addressed:

- The cumulative risk increase that would be computed if the unreliability of those SSCs were assumed to simultaneously increase by that factor. That is, the factor used can not lead to exceeding the quantitative acceptance guidelines of Reg. Guide 1.174.
- The ability of a monitoring program to detect a change of that factor. This includes consideration of currently expected number of failures for the number of demands/hours of operation and the expected number of failures for the expected future number of demands/hours of operation for the population of SSCs that are expected to be classified as LSS. Standard practices used for setting performance criteria based on failures under the maintenance rule are applicable.

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.

Section 12 of this document addresses considerations for maintaining the validity of the sensitivity study following initial categorization.

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 F-V 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 LSS 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

Rule Reference:

The guidance in this section addresses §50.69(c)(2). See Section 2 for further information.

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, engineering (e.g., design, systems, electrical, I&C including information technology, nuclear risk management), industry operating experience, licensing, 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. 10 CFR 50.69(c)(2) discusses the makeup of the IDP, and specifies categories of expertise, which are listed in the bullets below. 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
- Safety analysis
- Systems Engineering,
- 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 panel may be staffed with additional disciplines, such as licensing, if desired.

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 categorizations 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 safety significant 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:

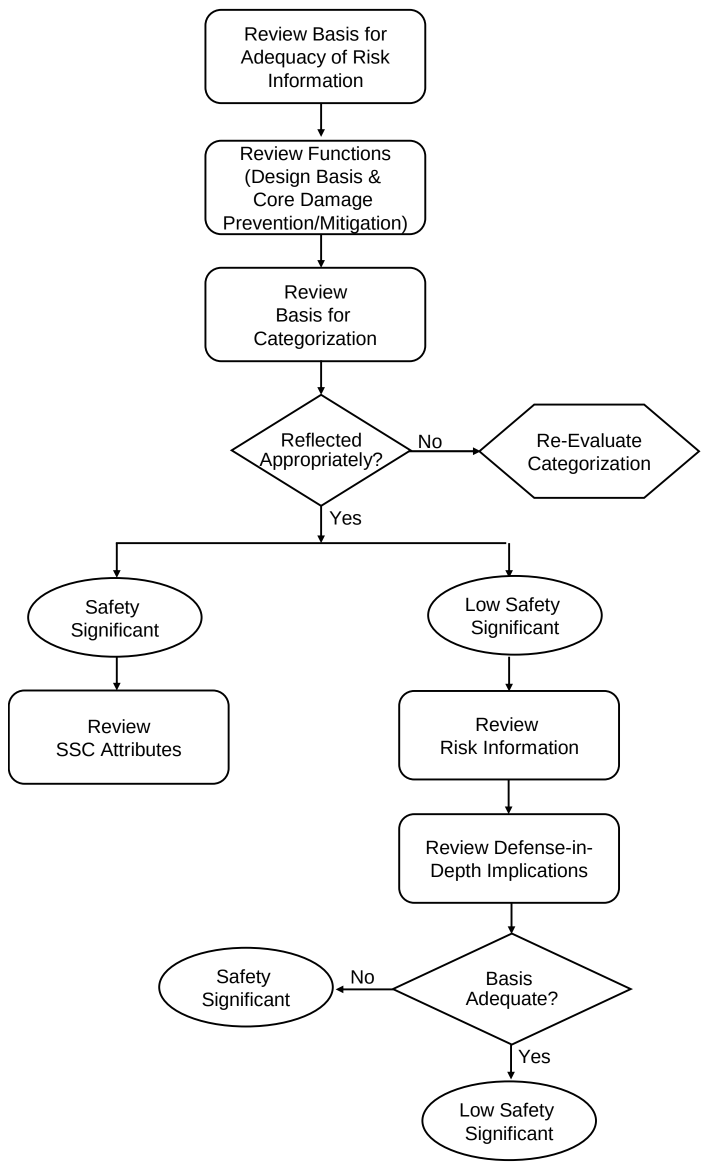
- the designated chairman, panel members, and panel alternates;
- required training and qualifications for the chairman, members, and alternates;
- requirements for a quorum, attendance records, agendas, and meeting minutes;
- the decision-making process;
- documentation and resolution of differing opinions; 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 of each function in the plant-specific risk analyses and defense-in-depth, is provided to the IDP for review. The overall categorization process to be used by the IDP is shown in Figure 9-1.

Figure 9-1

IDP PROCESS



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 IDP should conclude that the risk information is adequate to support categorization of the selected system. The appropriateness of the manner in which the function/SSC has been reflected should be judged based on the scope of functions considered and the manner in which the risk information incorporates those functions. If the IDP determines that the function/SSC 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.

The IDP review of the categorization of the functions/SSCs does not need to include the verification that all of the SSCs mapped to that function are appropriate. The IDP approval of the categorization of system functions, based on the coarse mapping of components to system functions, would be used to define the safety significance of each SSC as described in Section 10. Thus, if a system function is found to be safety-significant by the IDP, then all components required to support that function would initially be considered safety-significant.

If a more detailed categorization of the SSCs associated with a safety-significant function is performed after the initial IDP, then the basis for that re-categorization must be considered in a follow-up IDP session. 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.

9.2.1 Review of Safety-significant Functions/SSCs

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 functions/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 beyond design basis attributes that were identified as important to the core damage prevention and mitigation functions of the SSC. The identification of the critical attributes is important because they provide information to the treatment activity implementers.

9.2.2 Review of Safety Related Low Safety-Significant Functions/SSCs

The IDP's role for these functions is to perform a risk-informed assessment of the function/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 LSS, the IDP should determine whether these functions/SSCs are not implicitly depended upon to maintain safe shutdown capability, prevention of core damage and maintenance of containment integrity. In making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions (reactivity control, core cooling, heat sink, and RCS inventory), considering plant design features and operator actions. Specifically, the IDP should consider whether:

1. Failure of the active function/SSC will not directly cause an initiating event that was originally screened out of the PRA based on anticipated low frequency of occurrence. (Note: The risk categorization process evaluates the impact of initiating events modeled in the PRA).

2. Failure of the active function/SSC will not cause a loss of reactor coolant pressure boundary integrity resulting in leakage beyond normal makeup capability.

3. Failure of the active function/SSC will not adversely affect the defense-in-depth remaining to perform the function. This is evaluated by confirming that failure of an active function / SSC will not directly or indirectly (e.g., through spatial effects) fail a basic safety function. This applies to any function/SSC under consideration, including functions/SSCs that are assumed to be inherently reliable (e.g., piping and tanks) or those that may not be explicitly modeled in the PRA (e.g., room cooling systems and instrumentation and control systems).

4. The active function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient. This also applies to instrumentation and other equipment associated with the required actions.

5. The active function/SSC is not called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions.

6. Failure of the active function/SSC will not prevent the plant from reaching or maintaining safe shutdown conditions; and the active function/SSC is not significant to safety during mode changes or shutdown. The IDP should assume that the plant would be unable to reach or maintain safe shutdown conditions if a function/SSC failure results in the need for actions outside of plant procedures or available backup functions/SSCs.

7. Failure of the active function/SSC that acts as a barrier to fission product release during plant operation or during severe accidents would not result in the implementation of off-site radiological protective actions.

Review Defense-In-Depth Implications

When categorizing a function/SSC as LSS, the IDP should consider whether the defense-in-depth philosophy is maintained. Defense-in-depth is maintained if:

1. Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of consequences of an offsite release.

2. There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.

3. System redundancy, independence, and diversity are preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters.

4. Potential for common cause failures is taken into account in the risk analysis categorization.

5. The overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure that no significant increase in risk would occur.

Note: The above considerations are implicit in the risk categorization process. Normally, the IDP would be informed that the process has been followed and the above considerations met. However, the IDP may raise the above issues on a proposed categorization for a function/SSC if they believe there are unique circumstances, or as they deem appropriate.

If the IDP concludes that the categorization of the function/SSC as low safety-significant is not justified, based on the risk review or the defense in depth review, then the IDP can re-categorize the SSC to RISC-1. 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.

Review Safety Margin Implications

Because the only requirements that are relaxed for LSS 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.

9.2.3 Review of Non-Safety-Related LSS Functions/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 assure 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 LSS 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 assure that any core damage prevention and mitigation attributes that the IDP felt were significant are included in future treatment.

10 SSC CATEGORIZATION

Rule Reference:

The guidance in this section, in addition to the guidance in Section 9, addresses §50.69(c)(2). See Section 2 for further information.

10.1 Coarse SSC Categorization

After the IDP approves the categorization of system functions, then the initial coarse mapping of components to system functions is used to define the safety significance of each SSC. Thus, if a system function is found to be safety-significant by the IDP, then all components that are required for the system function should be considered safety-significant . In some cases, components may support both safety-significant and LSS system functions. In these cases, if the SSC is required for <u>any</u> safety-significant system function, then it should be considered safety-significant. Likewise, if <u>all</u> system functions supported by the SSC are LSS, then the SSC can be considered LSS.

For some systems or system functions, the SSC categorization based on the coarse mapping may provide adequate benefits to the licensee. In other cases, this approach may be too conservative, so a more detailed categorization may be utilized as discussed in Section 10.2.

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 supporting a 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 selected SSCs to a lower classification based on the attributes of the function that the SSC supports. 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 supporting a safety-significant function 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 supporting a safety-significant function would include, but are not limited to:

  - There is no credible 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.),
  - A 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 pathway 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.).

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 LSS based on the above considerations), the new categorization and its basis should be presented to another session of the IDP to be re-categorized using the same rigor as described in Section 9. If the SSCs being considered for re-categorization to a lower classification are modeled in the PRA, then the risk sensitivity described in Section 5 would need to be completed prior to presentation to the IDP.

11 PROGRAM DOCUMENTATION AND CHANGE CONTROL

Rule Reference:

The guidance in this section addresses §50.69(f).

§50.69(f) states the following:

(f) *Program documentation, change control and records.*

(1) The licensee or applicant shall document the basis for its categorization of any SSC under paragraph (c) of this section before removing any requirements under § 50.69(b)(1) for those SSCs.

(2) Following implementation of this section, licensees and applicants shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with § 50.71(e).

(3) When a licensee first implements this section for a SSC, changes to the FSAR for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.59 evaluation of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met.

(4) When a licensee first implements this section for a SSC, changes to the quality assurance plan for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.54(a) review of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the quality assurance plan may be made if the requirements of this section and § 50.54(a) continue to be met.

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 LSS.

For the purposes of this guidance, initial implementation refers to the first application of the 10 CFR 50.69 rule to a particular system. This may be at the time the first system(s) are categorized under 10 CFR 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. However, licensees should ensure that any design basis commitments continue to be maintained (Reference 10 CFR 50.2, NRC Regulatory Guide 1.186, and NEI 97-04, Rev 1 for definition of and guidance on design basis).

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

Rule Reference: 10 CFR 50.69(b)(2)(iv), (d)(1) and (e). 50.69 (d)(1) and (e) are quoted below:

10 CFR 50.69 (d) *Alternative treatment requirements.*

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

10 CFR 50.69 (e) *Feedback and process adjustment.*

> *(1) RISC-1, RISC-2, RISC-3 and RISC-4 SSCs. The licensee shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and SSC categorization and treatment processes. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.*
>
> *(2) RISC-1 and RISC-2 SSCs. The licensee shall monitor the performance of RISC-1 and RISC-2 SSCs. The licensee shall make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid.*
>
> *(3) RISC-3 SSCs. The licensee shall consider data collected in § 50.69(d)(2)(i) for RISC-3 SSCs to determine if there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations conducted to satisfy § 50.69(c)(1)(iv). The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization process and results are maintained valid.*

12.1 The following guidance is provided relative to paragraph 10 CFR 50.69(e)(1) above, for RISC-1, RISC-2, RISC-3, and RISC-4 SSCs

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review should be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information should be performed during the normally scheduled periodic review cycle.

Scheduled periodic reviews (e.g., once per two fuel cycles in a unit) should evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information[4] and the categorization process should be updated. This review should include:

- A review of plant modifications since the last review that could impact the SSC categorization
- A review of plant specific operating experience that could impact the SSC categorization,
- A review of the impact of the updated risk information on the categorization process results
- A review of the importance measures used for screening in the categorization process[5].
- An update of the risk sensitivity study performed for the categorization

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded[6], a review of the SSC categorization should be performed. It is expected that risk information upgrades would normally be timed such that the upgrade would coincide with the normal periodic review schedule. However, in the case that the upgrade was performed on a separate schedule, then the review should be performed in a timely manner, and should include similar considerations as those listed above for the periodic reviews.

In most cases, the categorization would be expected to be unaffected by changes in the plant-specific risk information. However, in some instances, an updated PRA model 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 SSC 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

[4] If multiple PRAs or analyses have been used to support the categorization process, the update may be limited to the specific risk information that has been determined to have changed in a manner that would affect the categorization process

[5] 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.

[6] PRA upgrade, as defined in ASME RA-S-2002, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant applications”, is as follows: *The incorporation into a PRA model of a new methodology or significant changes in scope or capability. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification and truncation, or new treatment of common cause failure.* NOTE- This means that adoption of an SPRA over a SMA would be an upgrade that requires an assessment.

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 plant design 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 based on the IDP process described in Section 9.

12.2 The following guidance applies to phased implementation:

In addition to the above considerations for periodic review, a planned and phased implementation of SSC categorization over several years could result in later SSC categorization activities impacting earlier SSC categorization schemes. Thus, a review of the impact of the current categorization activity on previous categorizations should be performed. A determination needs to be made whether the integrated sensitivity study or the defense in depth implication 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. This review of previous categorization may be focused to those SSCs affected by the categorization of additional functions, and does not obviate or replace the periodic review discussed in 12.1 above.

12.3 The following guidance is provided relative to paragraphs 10 CFR 50.69(d)(1) and 10 CFR 50.69(e)(2) above, for RISC-1 and RISC-2 SSCs

For initial implementation, paragraph 50.69(d)(1) is met through verification of PRA technical adequacy, as addressed through 50.69(b)(2)(ii). This ensures that a valid basis exists for the RISC-1 and RISC-2 performance credited in the categorization process. For implementation going forward, the provisions of 50.69(d)(1) and (e)(2) are met in a performance-based manner through monitoring, feedback, and updates to the PRA and/or the categorization results. RISC-1 and RISC-2 SSCs can be monitored in the same manner as they are monitored under 10 CFR 50.65, the Maintenance Rule, with the following clarifications:

a. The monitoring should address all functional failures, not just maintenance preventable functional failures.

b. The scoping requirements of the maintenance rule would be expected to envelop practically all RISC-1 and RISC-2 SSCs. However, to the extent that any of these SSCs are not in the maintenance rule scope, appropriate monitoring requirements should be developed for those SSCs.

As appropriate, the results of this monitoring should be used to determine if adjustments to the categorization assumptions, or treatment processes for RISC-1 and RISC-2 SSCs, are necessary.

12.4 The following guidance is provided relative to paragraph 10 CFR 50.69(e)(3) above, for RISC-3 SSCs

Paragraph §50.69(d)(2)(i) *Inspection and Testing*, states that periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions. Data obtained from this testing is used to satisfy the provisions of §50.69(e)(3).

Paragraph §50.69(d)(2)(ii), *Corrective Action*, states that conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition. The primary intent of this provision is to address the possible effects of potential common cause failures and degradation mechanisms following implementation, as discussed in paragraph 50.69(b)(2)(iv).

Common cause failures are an important consideration for implementation of §50.69. Common cause failures are defined as the simultaneous failure of more than one SSC to perform its function, due to the same cause (design, maintenance, environment, etc). Common cause failure is of particular interest for standby equipment, since normal operation may not reveal the failures until the function of the equipment is demanded by an initiating event, or during testing. While the §50.69 process maintains important defenses against common cause failure, it is possible that common cause failure rates could be affected through changes in special treatment of RISC-3 SSCs. Section 8 of this document discusses how various elements of the risk categorization (base PRA model requirements, common cause risk importance measures (RAW and FV), defense-in-depth evaluation, and integrated risk sensitivity study) ensures that the potential for common cause failures for RISC-3 SSCs is appropriately considered. In addition to the categorization process itself, the requirements of the rule for RISC-3 treatment, including test and inspection (§50.69(d)(2)(i)), periodic evaluation (§50.69(e)) and corrective action (§50.69(d)(2)(ii)), provide important defenses against the potential for common cause failures going undetected.

Performance monitoring of RISC-3 SSCs, as required by 10 CFR 50.69(e)(3), is established to provide assurance that potential increases in failure rates will be detected and addressed before reaching the rate assumed in the integrated sensitivity study. Since implementation of §50.69 would allow RISC-3 SSCs to be procured with reduced special treatment, and used in multiple systems, it is important to be aware of inter-system common cause failure potential. As a means to monitor equipment performance changes, failures of RISC-3 SSCs are identified and tracked in a corrective action program. As part of the corrective action program, failures of RISC-3 SSCs are reviewed to determine the extent of condition (i.e., whether this failure is indicative of a potential common cause failure). For the purposes of assessing data from the corrective action program, failures should be assessed for groups of like component types (e.g., motor operated valves, air operated valves, motor-driven pumps, etc). The intent of the periodic review is twofold: first, to ensure that the failure rate of RISC-3 SSCs in a given time period has not unacceptably increased due to the changes in treatment. The periodic review validates that the rate of RISC-3 SSC equipment failures has not increased by a factor greater than that used in the integrated risk sensitivity study. Second, the review of component group failure data is performed to detect the occurrence of potential inter-system common cause failures, and to allow timely corrective action if necessary, as required by §50.69(d)(2)(ii). Since most RISC-3 components have low failure rates, noted increases to these rates are most readily detected through grouping of components. If failure rate increases are noted, attention should be focused to common treatment changes to groups of components to ensure that the potential for inter-system common cause failure remains low. This corrective action review should also consider previous component performance history.

This review of failure experience accounts for any changes in test frequencies, routine demands and exposure times, as appropriate. This can be accomplished by proactively assessing the documented failures in a given group of SSCs, and comparing the number of failures documented in the current review period against failures in previous periods, accounting for changes in treatment. If the number of failures for a group of SSCs exceeds a factor of two increase over the expected number of failures, a potential adverse trend is identified requiring further assessment. The factor of two is selected so to assure an assessment is initiated prior to exceeding the factor used in the risk sensitivity study (e.g., a factor of 3 to 5). The licensee should take the appropriate actions, (which could include changes in treatment or categorization), to preclude reaching unacceptable performance.

13 REFERENCES

1. 10 CFR50.69, Final Rule, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors*, November 22, 2004.
2. EPRI TR-105396, *PSA Applications Guide*, August 1995
3. NRC Regulatory Guide 1.174, Revision 1, *An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*, November 2002
4. NRC SECY 99-256, *Rulemaking Plan For Risk-Informing Special Treatment Requirements*, October 29, 1999
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 Guide 1.175, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing, August 1998
8. NRC Regulatory Guide 1.176, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance , August 1998
9. NRC Regulatory Guide 1.177, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, August 1998
10. NRC Regulatory Guide 1.178, Revision 1, An Approach For Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping, September 2003
11. NRC Regulatory Guide 1.200, *An Approach for Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities*, Issued For Trial Use, February 2004.
12. Nuclear Energy Institute, "NEI 00-02, Revision A3, *Probabilistic Risk Assessment (PRA) Peer Review Process Guidance*," March 20, 2000
13. NEI 96-07, *Guidelines for 10 CFR 50.59 Safety Evaluations*
14. NEI 97-04, Revision 1, *Design Bases Program Guidelines*
15. NEI 98-03, *Guidelines for Updating Final Safety Analysis Reports*
16. 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."*
17. ASME Code Case, N-660, *"Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," July 2002.*
18. ASME RA-S-2002, *Standard for Probabilistic Risk Assessments for Nuclear Power Plant Applications*, April 5, 2002
19. EPRI TR- 008905, *Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization*, June 2003.

APPENDIX A

10 CFR 50.69 Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors

(a) *Definitions.*

“ Risk-Informed Safety Class (RISC)-1 structures, systems, and components (SSCs)” means safety-related SSCs that perform safety-significant functions.

“ Risk-Informed Safety Class (RISC)-2 structures, systems and components (SSCs)” means nonsafety-related SSCs that perform safety-significant functions.

“ Risk-Informed Safety Class (RISC)-3 structures, systems and components (SSCs)” means safety-related SSCs that perform low safety-significant functions.

“ Risk-Informed Safety Class (RISC)-4 structures, systems and components (SSCs)” means nonsafety-related SSCs that perform low safety-significant functions.

“ Safety-significant function” means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

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

(1) A holder of a license to operate a light water reactor (LWR) nuclear power plant under this part; a holder of a renewed LWR license under Part 54 of this chapter; an applicant for a construction permit or operating license under this part; or an applicant for a design approval, a combined license, or manufacturing license under Part 52 of this chapter’ may voluntarily comply with the requirements in this section as an alternative to compliance with the following requirements for RISC- 3 and RISC- 4 SSCs:

(i) 10 CFR Part 21.

(ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR Part 50.

(iii) 10 CFR 50.49.

(iv) 10 CFR 50.55(e).

(v) The inservice testing requirements in 10 CFR 50.55a(f); the inservice inspection, and repair and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in section 4.3 and 4.4 of IEEE 279, and sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h).

(vi) 10 CFR 50.65, except for paragraph (a)(4).

(vii) 10 CFR 50.72.

(viii) 10 CFR 50.73.

(ix) Appendix B to 10 CFR Part 50.

(x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, for penetrations and valves meeting the following criteria:

(A) Containment penetrations that are either 1- inch nominal size or less, or continuously pressurized.

(B) Containment isolation valves that meet one or more of the following criteria:

(1) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(2) The valve is normally closed and in a physically closed, water-filled system;

(3) The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and that is not connected to the reactor coolant pressure boundary; or

(4) The valve is 1-inch nominal size or less.

(xi) Appendix A to Part 100, sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake.

(2) A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:

(i) A description of the process for categorization of RISC-1, RISC-2, RISC-3 and RISC-4 SSCs.

(ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant- specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.

(iii) Results of the PRA review process conducted to meet § 50.69 (c)(1)(i).

(iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69 (c)(1)(iv). The evaluations shall include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

(3) The Commission will approve a licensee's implementation of this section if it determines that the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs satisfies the requirements of § 50.69(c) by issuing a license amendment approving the licensee's use of this section.

(4) An applicant for a license voluntarily choosing to implement this section shall include the information in § 50.69(b)(2) as part of application. The Commission will approve an applicant's implementation of this section if it determines that the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs satisfies the requirements of § 50.69(c).

(c) SSC Categorization Process.

(1) SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines whether an SSC performs one or more safety- significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth.

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § 50.69(b)(1) and § 50.69(d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

(2) The SSCs must be categorized by an Integrated Decision- Making Panel (IDP) staffed with expert, plant- knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

(d) *Alternative treatment requirements.*

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

(2) RISC-3 SSCs. The licensee or applicant shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC-3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action, shall be provided for RISC-3 SSCs.

(i) *Inspection and Testing*. Periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions; and

(ii) *Corrective Action*. Conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

(e) *Feedback and process adjustment.*

(1) RISC-1, RISC-2, RISC-3 and RISC-4 SSCs. The licensee shall review changes to the plant, operational practices, applicable industry operational experience, and, as appropriate, update the PRA and SSC categorization. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.

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

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

(f) *Program documentation, change control and records.*

(1) The licensee or applicant shall document the basis for its categorization of any SSC under paragraph (c) of this section before removing any requirements under § 50.69(b)(1) for those SSCs.

(2) Following implementation of this section, licensees and applicants shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with § 50.71(e).

(3) When a licensee first implements this section for a SSC, changes to the FSAR for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.59 evaluation of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the FSAR, may be made if the requirements of this section and § 50.59 continue to be met.

(4) When a licensee first implements this section for a SSC, changes to the quality assurance plan for the implementation of the changes in accordance with § 50.69(d) need not include a supporting § 50.54(a) review of the changes directly related to implementation. Thereafter, changes to the programs and procedures for implementation of § 50.69(d), as described in the quality assurance plan may be made if the requirements of this section and § 50.54(a) continue to be met.

(g) *Reporting.* The licensee shall submit a licensee event report under § 50.73(b) for any event or condition that would have prevented RISC-1 or RISC-2 SSCs from performing a safety-significant function.

APPENDIX B

GLOSSARY OF SELECTED TERMS

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

Common cause failure (CCF) **–** A failure of two or more components during a short period of time as result of a single shared cause (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

Core damage **–** Uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

Core damage frequency (CDF) **–** Expected number of core damage events per unit of time (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

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

Design bases - means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. (Ref: 10 CFR 50.2)

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

Dependency **–** Requirement external to an item and upon which its function depends and is associated with dependent events that are determined by, influenced by, or correlated to other events or occurrences (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

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

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

Large early release **–** The rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

Large early release frequency (LERF) **–** Expected number of large early releases per unit of time (Ref: ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications)

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

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 **-** Risk encompasses what can happen (scenario), its likelihood (probability), and its level of damage (consequences). (Ref: NUMARC 93-01, Rev 2, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants)

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

Safety-related structures, systems and components - means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary
(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

(Ref: 10 CFR 50.2)

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

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

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