UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 2000

MEMORANDUM TO: Cynthia A. Carpenter, Chief
Generic Issues, Environmental, Financial
and Rulemaking Branch
Division of Regulatory Improvement Programs

FROM: Anthony W. Markley, Program Manager
Generic Issues, Environmental, Financial
and Rulemaking Branch
Division of Regulatory Improvement Programs

SWest for

SUBJECT: SUMMARY OF PUBLIC MEETING HELD ON MARCH 30, 2000 TO DISCUSS SPECIAL TREATMENT REQUIREMENTS

On March 30, 2000, the Office of Nuclear Reactor Regulation (NRR) held a public meeting with the Nuclear Energy Institute (NEI) and other interested stakeholders to discuss key issues involved with the development of a proposed rule for risk-informing the special treatment requirements of 10 CFR Part 50 (RIP-50). Representatives of the Office of Nuclear Regulatory Research, the American Society of Mechanical Engineers (ASME), a number of reactor licensees, consultants, and others also attended and participated in the meeting. Attachment 1 lists meeting participants. Attachment 2 contains the draft partial guideline for risk-informed categorization of structures, systems, and components e-mailed to the staff by NEI. Attachment 3 contains typical elements in a commercial work control program e-mailed to the staff by NEI. Attachment 4 provides a set of comments presented by American Society of Mechanical Engineers.

NEI described its continuing efforts to develop guidance for classifying the safety significance and risk importance of structures, systems, and components (risk-informed safety categories) and for implementing changes to the special treatment requirements. NEI provided a partial draft of its proposed guideline for risk-informed categorization and treatment of SSCs. While not complete, this document provides guidance and decision logic for risk-informed categorization of SSCs for appropriate levels of treatment. NEI indicated that this draft would be submitted for formal staff review by the end of April 2000. In the interim, NEI indicated that they would appreciate informal staff comments on this document. NEI also provided information on commercial quality work control and practices.

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NEI also discussed its efforts to solicit pilot plants. The Boiling Water Reactor Owners Group (BWROG) has received funding and expects final approval in May 2000. The BWROG will use a two-system approach with a lead plant. The effort should culminate in a BWROG document and a boiler plate exemption package by the end of calender year (CY) 2000. This effort would be broadened to cover more plant systems with applicability to the rest of the BWR fleet in CY 2001. The Westinghouse Owners Group (WOG) is still evaluating the extent to which it may participate in the pilot program.

Beyond the owners groups, NEI indicated that it is the industry's general view that better understanding of how treatment will be handled and favorable progress on the staff's review of the South Texas Project (STP) exemption requests will be needed before licensees commit to "whole-plant" pilot activities. The staff stressed the importance of integrating pilot plant activities into the overall plan for developing the proposed rule and that the pilot activities should be conducted within a time frame that will support the rulemaking. The staff also noted the importance of applying the lessons-learned from the pilot activities to the proposed rulemaking.

The staff and NEI agreed to continue to hold regular meetings and have scheduled the next meeting for April 20, 2000. At that meeting, NEI will give the staff a final draft of its guidance document and will further discuss potential pilot plant activities.

Attachments: as stated

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Attachments: as stated

List of Attendees

Name	Organization	Phone Number
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R. Young	NRC	301-415-2852
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3/29/00 DRAFT

Industry Guideline for Risk-Informed Categorization And Treatment of Structures, Systems, and Components

Table of Contents

Attachment 2

1.0 Background

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

Since the inception of this regulatory basis, over 2500 reactor years of operation have been accumulated in the US (over 6000 reactor years worldwide), with a corresponding body of data relative to actual transients, accidents, and plant equipment performance. Further, each US plant has performed a probabilistic risk analysis (PRA), which uses these data, and models a large number of potential accident sequences (including sequences not considered in the deterministic regulatory basis) to estimate the overall risk from plant operation. PRAs describe risk in terms of the frequency of reactor core damage and/or significant offsite release. Insights from PRAs reveal that certain plant equipment important to the deterministic regulatory basis is of little significance to risk, and, conversely, certain plant equipment is important to risk, but was not included in the deterministic regulatory basis.

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

1.1 Regulatory Reform Initiative

Current NRC regulations establish that plant equipment necessary to meet the deterministic regulatory basis is categorized as "safety related", and is subject to a

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

broad set of "special treatment[1]" regulations to assure performance capability, awareness and involvement of the regulator, and many other factors. Other plant equipment is categorized as "non safety related", and is not subject to special regulatory treatment. The objective of regulatory reform is to modify the scope of equipment subject to special regulatory treatment in light of risk insights from PRAs and plant operation. This will result in reduction of special treatment provisions for safety related equipment with low risk significance, and addition of regulatory provisions for non safety related equipment with high risk significance.

NRC has proposed a new regulation, 10 CFR 50.69, that would provide an option for licensees to implement a risk-informed approach for regulations that establish special treatment requirements for plant structures, systems and components (SSCs). Table 1.1 lists the special treatment regulations that would be subject to the optional risk-informed approach. 10 CFR 50.69 would define four categories of SSCs, based on existing safety classification and risk significance, and establish special treatment provisions as a function of the categorization. The special treatment regulations in Table 1.1 would not in themselves be changed. However, the scope of applicability, and the manner in which the special treatment provisions are implemented, would be revised as defined in 10 CFR 50.69. Proposed Appendix T to Part 50 would provide additional regulatory requirements for the categorization process.

The decision to adopt a risk-informed approach for categorizing structures, systems and components is voluntary. Each licensee will make its determination on whether to adopt a risk-informed approach to regulation based on the estimated benefit.

The NRC rulemaking plan does not replace the existing "safety related" and "non safety related" classifications, because this would require rulemaking to individual special treatment regulations. Rather, 10 CFR 50.69 would provide that the each existing classification category can be divided into two categories, based on high or low risk significance. The categorization is depicted below.

Risk Informed Safety Classifications (RISC)

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

RISC 1 Existing classification: Safety Related Risk Significance: High	**RISC 2** Existing classification: Non Safety Related Risk Significance: High
RISC 3 Existing classification: Safety Related Risk Significance: Low	**RISC 4** Existing classification: Non Safety Related Risk Significance: Low

The application of special treatment regulations would be a function of the above categorization. Regulatory requirements would apply for all categories except RISC 4. The existing special treatment provisions for RISC 1 equipment would be maintained. RISC 2 equipment would be subject to new regulatory controls. RISC 3 equipment would be subject to reduction of existing regulatory controls, however, it is not intended that such SSCs could be removed from the facility, or have their functional capability lost.

Regulatory requirements for equipment in RISC 2 and RISC 3 would, to the extent achievable, use a performance based approach, similar to that of the existing maintenance rule. This approach uses performance monitoring, rather than special treatment, to ensure equipment reliability. To the extent that monitoring could not address the important safety function of equipment in RISC 2, special treatment provisions would be limited to the attributes that directly relate to the safety function. Otherwise, standard industrial practices would apply to equipment outside of RISC 1.

Treatment of SSCs

RISC 1 Maintain existing special treatment requirements per regulations of Table 1.1	**RISC 2** Monitor risk significant attributes Control risk-significant attributes not addressable by monitoring Standard industrial treatment
RISC 3 Monitor performance Standard industrial treatment	**RISC 4** Standard industrial treatment

1.2 Categorization Pathways

The risk-informed classification scheme allocates each SSC in the plant to one of four classifications (RISC 1 – 4). Figure 1.1-1 Provides a graphical depiction of the classification pathways utilized in this process. The existing safety related components in the plant are classified either via pathway 1 to RISC-1 (for safety significant SSCs) or via pathway 2 to RISC-3 (for low safety significant SSCs). Pathway 1 is the default pathway for all safety related SSCs. That is, unless a compelling case can be made that the safety related SSC is low safety significant, then it is classified as RISC-1. In cases where a risk-informed process can demonstrate that the safety related SSC is of low safety significance, it is classified as RISC-3.

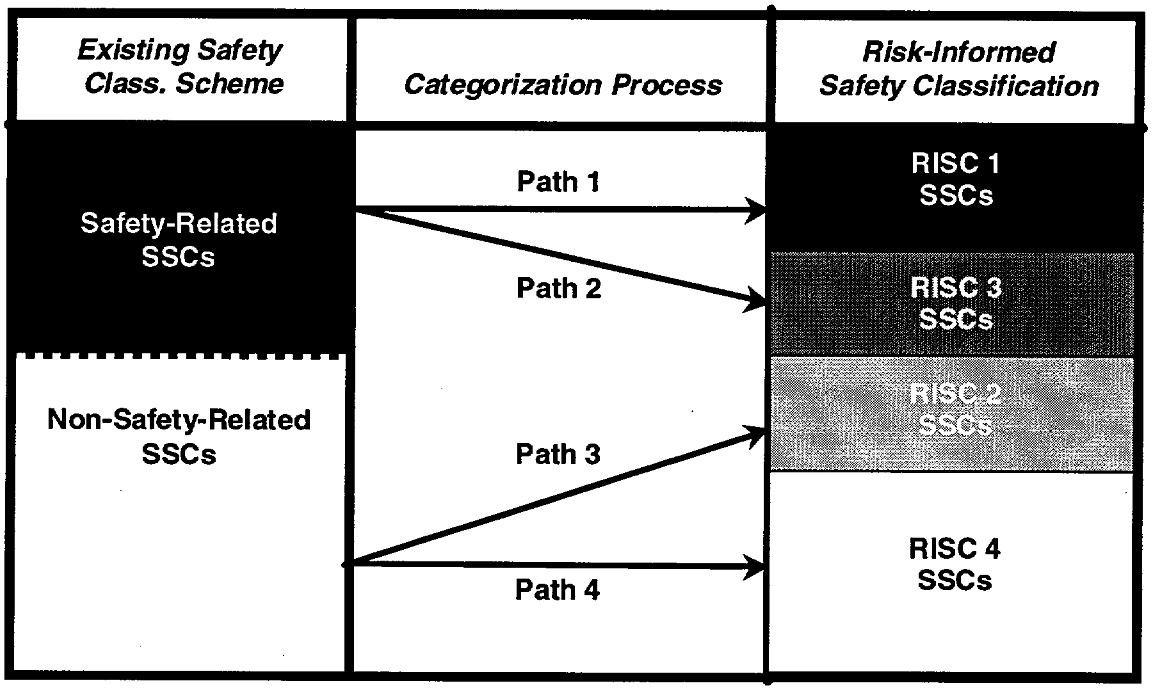
All other SSCs (non-safety related) are classified on either pathway 3 to RISC-2 (for safety significant SSCs) or via pathway 4 to RISC-4 (for low safety significant SSCs). In this case, pathway 4 is the default pathway for non-safety related SSCs. That is, unless a compelling case can be made that the non-safety related SSC is safety significant, then it is classified as RISC-4. In cases where a risk-informed process can demonstrate that the non-safety related SSC is safety significant, it is classified as RISC-2.

The only time that an SSC would move via another pathway would be if it was found to be misclassified initially. In that case, the licensee would have to first demonstrate that the initial classification was inappropriate; however, reclassification for this purpose is based on the existing deterministic licensing basis, does not involve use of risk insights, and is not a subject of this guidance.

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

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

Figure 1.1-1
CONCEPTUAL REPRESENTATION OF CLASSIFICATION PATHWAYS



1.3 Implementation Process

This document provides detailed implementation guidance for 10 CFR 50.69 and Appendix T. Plants that follow the guideline should be able to implement risk-informed regulation with minimal NRC review. Since this guidance is used to effect a change to the plant's licensing basis, it follows the principles of NRC Regulatory Guide 1.174, as follows:

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

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

The process for implementation involves four elements:

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

The first element involves determining the plant systems to which the revised approach would be applied. Plant systems that can impact PRA initiating events and accident mitigation are candidate systems for application of the process. Certain plant systems have regulatory requirements that have bases other than protection of public health and safety from potential reactor accidents (e.g., the radwaste processing system). These systems, and their associated regulatory

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

requirements, are not within the scope of the process.

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

The second principal activity is the categorization of the SSCs according to safety significance. Treatment requirements for SSCs will be dependent on this safety classification. This report establishes an integrated process which relies upon the insights from plant-specific risk analyses and other engineering and operating inputs for use in the categorization of SSCs.

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

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

1.4 References

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

Since 1991 the industry and the NRC has developed background documents and guidance for the application of risk-informed applications. Several of these documents had significant impact on the development of this guidance including:

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

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

Table 1.1

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

50.34, Contents of applications; technical information (FSAR)
50.36, Technical specifications
50.44, Combustible gas control
50.48, Fire protection
50.49, Environmental qualification
50.54(a)(3), Conditions of licenses (in reference to Quality Assurance Programs only)
50.55, Conditions of construction permits
50.55a, Codes and standards
50.59, Changes, tests and experiments
50.65, Monitoring effectiveness of maintenance
50.71(e), Maintenance of records, making of reports
50.72/50.73, Reporting
Appendix A, General Design Criteria
  GDC 1, Quality standards and records
  GDC 2, Design bases for protection against natural phenomena
  GDC 3, Fire protection
  GDC 4, Environmental and dynamic effects design bases
  GDC 37, Testing of emergency core cooling system
  GDC 40, Testing of containment heat removal system
  GDC 42, Inspection of containment atmosphere cleanup systems
  GDC 43, Testing of containment atmosphere cleanup systems
  GDC 45, Inspection of cooling water system
  GDC 46, Testing of cooling water system
Appendix B, Quality Assurance
Appendix J, Containment leakage
Appendix R, Fire Protection
Appendix S, Seismic
Part 21, Reporting of defects and noncompliance
Part 52, Advanced Reactors
Part 54, License Renewal
Part 100, Appendix A, Seismic

Section 2 - Categorization Process

2.1 GUIDING PRINCIPLES

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

The objective of this guidance is to establish the process and criteria for determining the SSCs that require special treatment. By defining the SSCs that require special treatment, those which do not require special treatment are identified by exception. The process and criteria are intended to be sufficiently clear and robust such that if a licensee's program meets the criteria there is not a need for prior NRC review and approval of the plant-specific program.

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

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

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

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

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

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

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

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

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

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

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

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

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

  4. <u>Any Increases In Core Damage Frequency Or Risk Should Be Small And Consistent With The Intent Of The Commission's Safety Goal Policy</u>

Statement - The re-classification and treatment process provides reasonable assurance that safety functions are maintained. Risk sensitivity studies will be used to demonstrate that no significant change in CDF and LERF will be expected.

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

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

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

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

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

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

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

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

2.2 SAFETY SIGNIFICANT ATTRIBUTES OF SSCs

One of the central concepts of the risk-informed safety categorization process is performance attributes. The risk-informed performance of many SSCs is the same (or similar) as that required in the design basis. At one time, it was expected that the design basis attributes would envelope all performance attributes. In many cases, this is true. For example, stroke times for valves are generally set based on conservative thermal hydraulic analyses which lead to performance requirements far in excess of those which a PRA would require (valves required to open in seconds when the PRA may indicate that minutes are available). In other cases, SSCs can have significantly different performance needs for severe accident mitigation. SSCs may be used in a unique manner or the conditions under which performance is desired may be more severe than the design basis considered. For example, pressurizer PORVs have a design basis to open to relieve primary system pressure. While that function may (or may not) be important in a PRA, another function not considered in the design basis is likely to be: open on demand to support bleed and feed cooling of the RCS in the event of loss of all secondary cooling.

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

Functional attributes can be broadly classified into four major categories:

- <u>SSC Function</u>

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

- <u>Performance Attributes</u>

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

- <u>Environmental Factors</u>

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

- Actuation Requirements

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

A list of possible performance and material attributes is provided for information in Appendix A.

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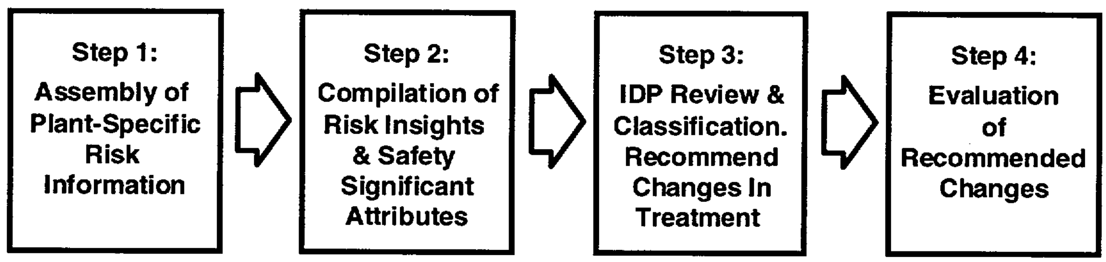
2.3 OVERALL APPROACH

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

Figure 2.3-1

Risk-Informed Classification Process



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

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

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

Determines:
- ✓ Specific Changes In Treatment
- ✓ Monitoring

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

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

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

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

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

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

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

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

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

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

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2.4 SPECIFIC GUIDANCE

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

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

2.4.1 Assembly of Plant-Specific Risk Information

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

2.4.1.1 Documentation Resources

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

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

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

Examples of resources which 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

2.4.1.2 Quality of PRA Information

At a minimum, a PRA modeling the internal initiating events at full power operations must be used to provide input to the IDP. The PRA must be capable of quantifying core damage frequency (CDF) and large early release frequency (LERF) and must reasonably reflect the as-built and as-operated plant. In general, the more applicable PRA information, the better. PRAs provide an integrated means to assess relative significance. In cases where applicable quantitative analyses are not available, the categorization process will generally identify more SSCs as safety significant than in cases where full scope PRAs are available.

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

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

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

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

- Use personnel qualified for the analysis.
- Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and

information used in the analyses (an independent peer review or certification program can be used as an important element in this process).

- Provide documentation and maintain records in accordance with accepted practices.

- Provide for an independent audit function to verify quality (an independent peer review or certification program can be used for this purpose).

- Use procedures that ensure appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decision-making is changed (e.g., licensee voluntary action) or determined to be in error.

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

2.4.2 Compilation of Risk Insights & Safety Significant Attributes

The compilation of risk insights and identification of safety significant attributes builds upon the plant-specific resources. An overview of the safety significance process is shown in Figure 2.4-1.

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

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

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

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

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

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

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

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

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

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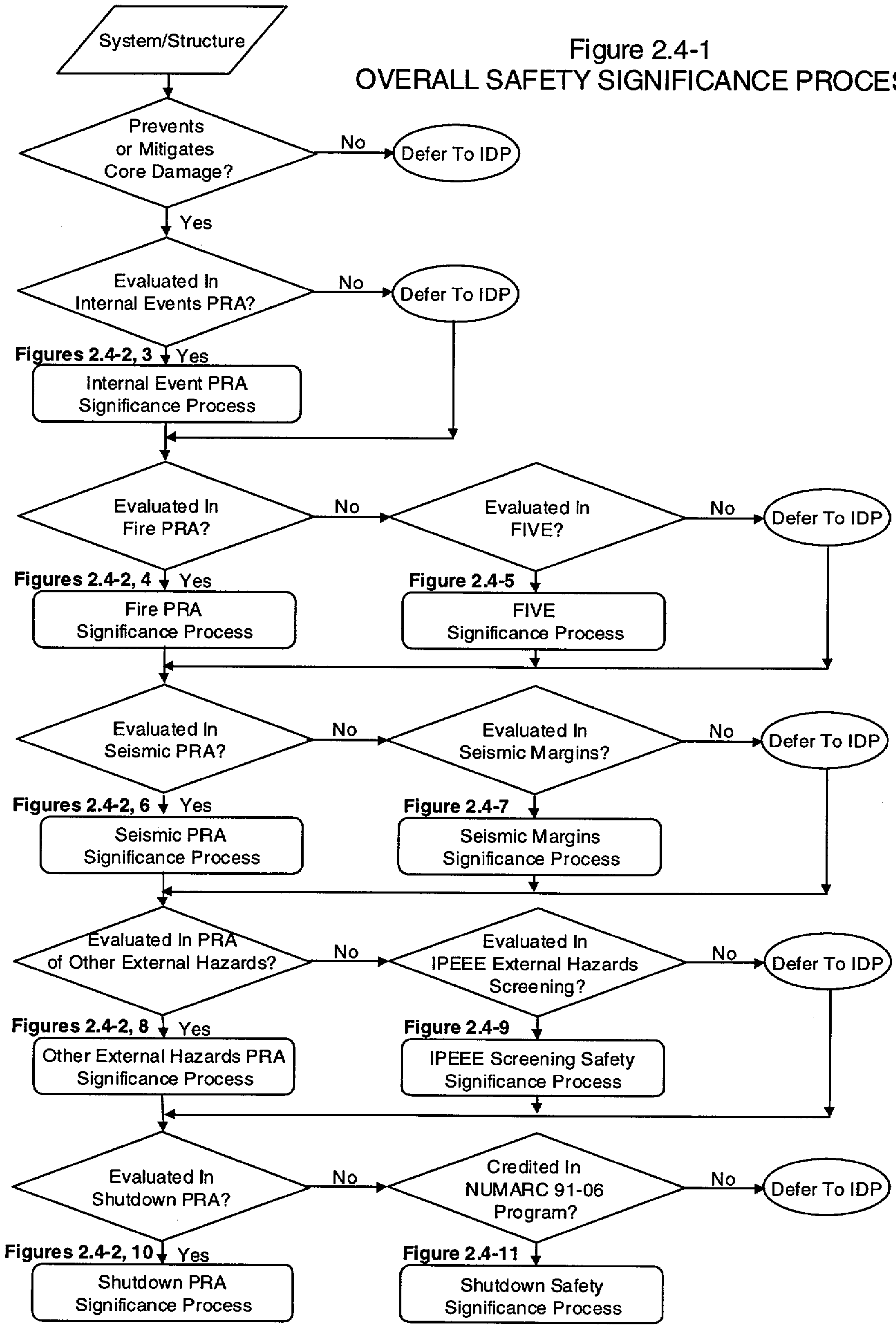
Figure 2.4-1
OVERALL SAFETY SIGNIFICANCE PROCESS



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

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

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

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

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

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

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

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

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

2.4.2.1 Internal Event Assessment

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

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

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

Components which do support a PRA function are evaluated using the risk importance process shown in Figure 2.4-3. 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. This process questions whether the SSC can directly cause a complicated initiating event which has a Fussell-Vesely importance greater than the criteria (0.005), then it is considered a candidate safety significant and the attributes which could influence that role as an initiating event are to be identified. A complicated initiating event is considered an event which trips the plant <u>and</u> 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 risk importance process utilizes two standard PRA importance measures, risk achievement worth (RAW) and Fussell-Vesely (F-V), as screening tools to identify candidate high safety significant SSCs. Risk reduction worth (RRW) is also an acceptable measure in place of Fussell-Vesely. The Fussell-Vesely criteria can be readily converted to RRW criteria. The Fussell-Vesely importance of a component is considered to be the sum of the F-V importances for the relevant failure modes of the component, including common cause failure. If a component does not have a common cause event to be included in the computation of importances, then an assessment should be made as to whether a common cause event should be added to the model. The RAW importance of a component is considered the maximum of the RAW values computed for basic events involving the component. In the case of RAW, the common cause event is not considered in the assessment of component risk significance. However, for RISC-3 SSCs, the common cause RAW can be used to target treatment activities which address common cause.

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

EXAMPLE IMPORTANCE SUMMARY

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

In cases where the core damage frequency is dominated by flooding, it is appropriate to break the evaluation of importance measures into two steps. The first step uses importance measures computed using the entire internal event 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, it is considered a candidate safety significant component and the safety significant attributes are to be identified. If the risk importance measure criteria are not met, then it is <u>not</u> automatically low safety significant. It must be evaluated as part of several sensitivity studies, determined to be low safety significant for all risk contributors and must be reviewed by the IDP. If the importance measures computed by the PRA tool do not indicate that a component meets the Fussell-Vesely or RAW criteria, then sensitivity studies are used to determine whether other conditions might lead to the component becoming safety significant. The recommended sensitivity studies for internal events PRA are identified in Table 2.4-1:

Table 2.4-1
<u>Sensitivity Studies For Internal Events PRA</u>

Sensitivity Study
• Increase all post accident human error probabilities by a factor of 10 or to a nominal value of 1E-3 (for small HEPs) Set all common cause failures to 0.0 Set all maintenance unavailability terms to 0.0 Increase all random failure probabilities (fail to start/open/run, etc.) by their associated error factor (e.g., 3 to 10) Others?? <still under consideration>

The sensitivity studies on human error rates, common cause failures and maintenance unavailabilities are performed to assure that assumptions of the PRA are not masking the importance of an SSC. The sensitivity addressing the variation of random failure probabilities is performed to assure that anticipated variations in individual SSC performance would be unlikely to change the classification.

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, it is a candidate for RISC-3. In this case the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

This risk importance process is performed for both CDF and LERF. In calculating the FV risk importance measure, it is recommended that a CDF (or LERF) cutset truncation level of at least five orders of magnitude below the baseline CDF (or LERF) value be used. For 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. When the RAW risk importance measure is calculated by a full re-solution of the plant PRA model, then the truncation level does not significantly affect the RAW calculations. In this case, a default truncation value of 1E-9 /yr seems reasonable. However, if a pre-solved set of cutsets is used to calculate RAWs, the truncation level should be set to a sufficiently low value so that all SSCs with RAW>2 are identified (e.g., cutoff of 1E-10 /yr or lower).

In cases where the component is safety related and found to be of low safety significance, it is appropriate to confirm that defense in depth is preserved. This discussion should include consideration of the events mitigated, the functions performed, the other systems which support those functions and the complement of other plant capabilities which can be relied upon to prevent core damage and large, early release. This assessment should consider both the level of defense in depth and to the frequency of the events being mitigated. The table below is an example of such an assessment:

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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 10^3-10^4 yr	LOCAs Other Design Basis Accidents				

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

In addition, the impact of common cause failures should also be assessed. If the safety related SSC is considered low safety significant, then the impact of common cause failure on CDF and LERF should be reviewed using the risk achievement worth (RAW) of the common cause event evaluated in the PRA. If the RAW of the common cause event is greater than 2, then the IDP should be notified that the SSC is common cause sensitive.

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

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

Figure 2.4-2
GENERALIZED SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN PRA

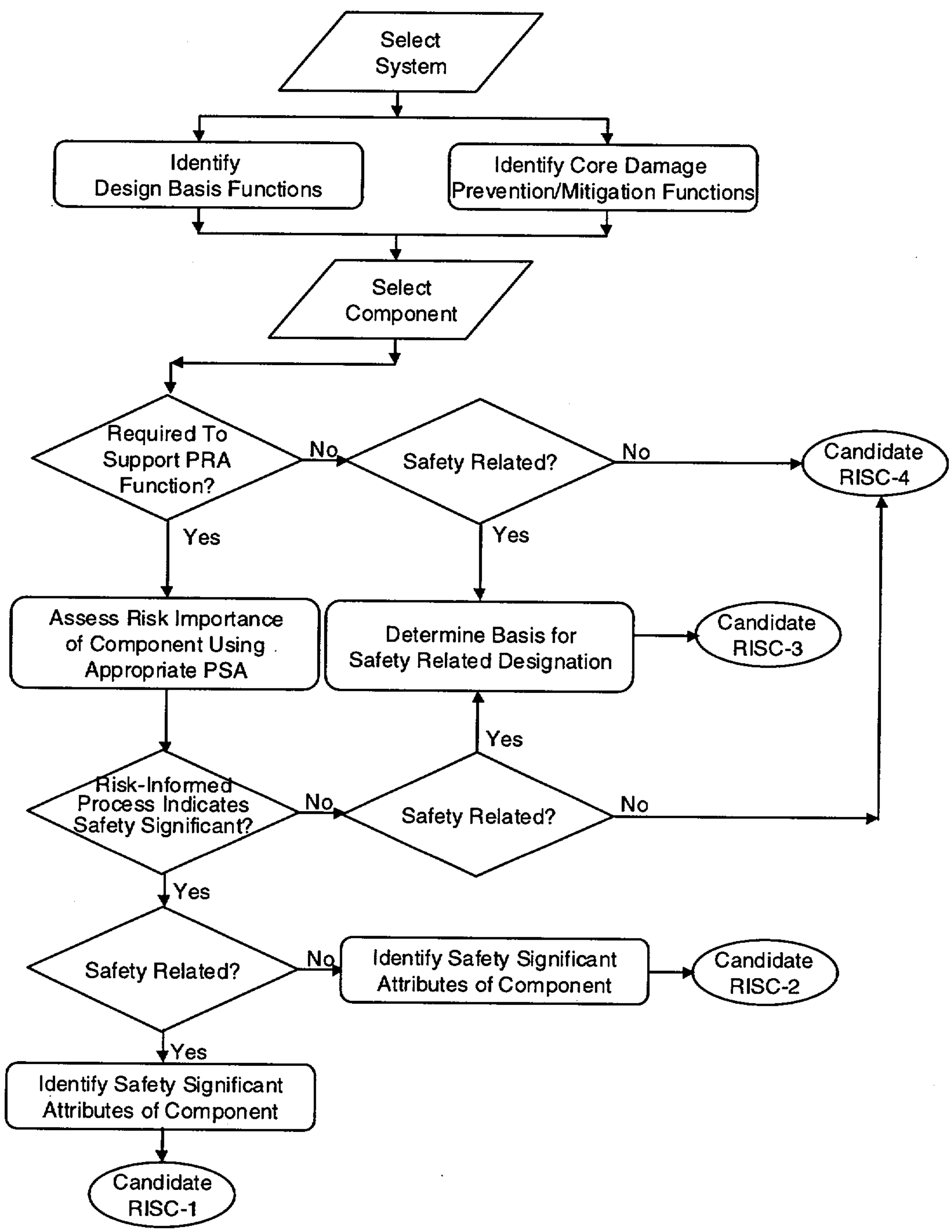

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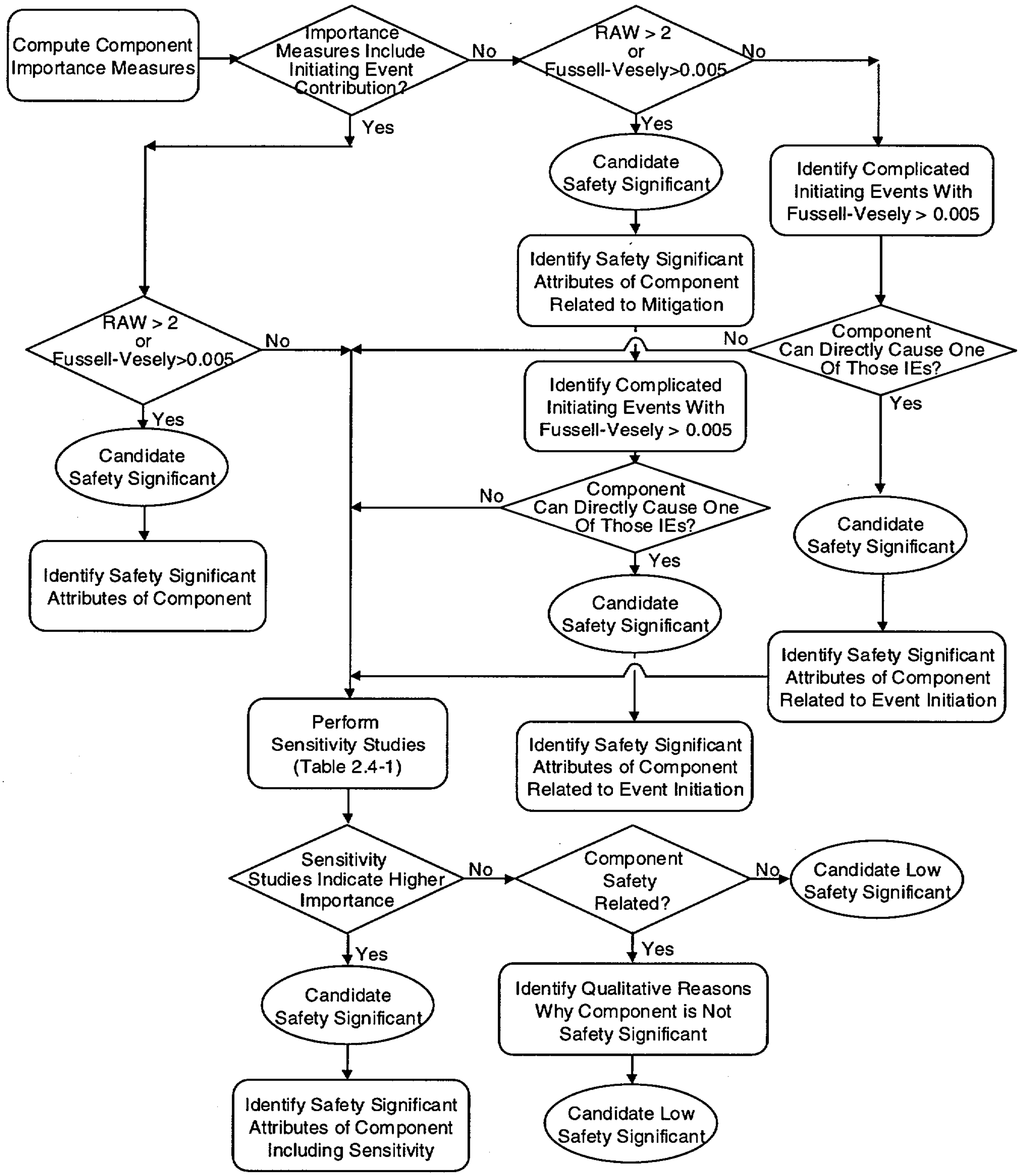
Figure 2.4-3
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN INTERNAL EVENTS AT-POWER PRAs



2.4.2.2 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 internal events PRA. This process is shown on Figures 2.4-2 and 2.4-4 and is discussed below. Plants which relied upon a FIVE analysis to assess fire risks for the IPEEE would use a modified process shown in Figure 2.4-5.

The generalized safety significance process for plants with a fire PRA is the same as the process for internal events. 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 and separately for the potential to initiate a fire. Aside from that small change, the process is the same as the internal event PRA process.

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

The recommended sensitivity studies for fire PRA are identified in Table 2.4-2:

Table 2.4-2
Sensitivity Studies For Fire PRA

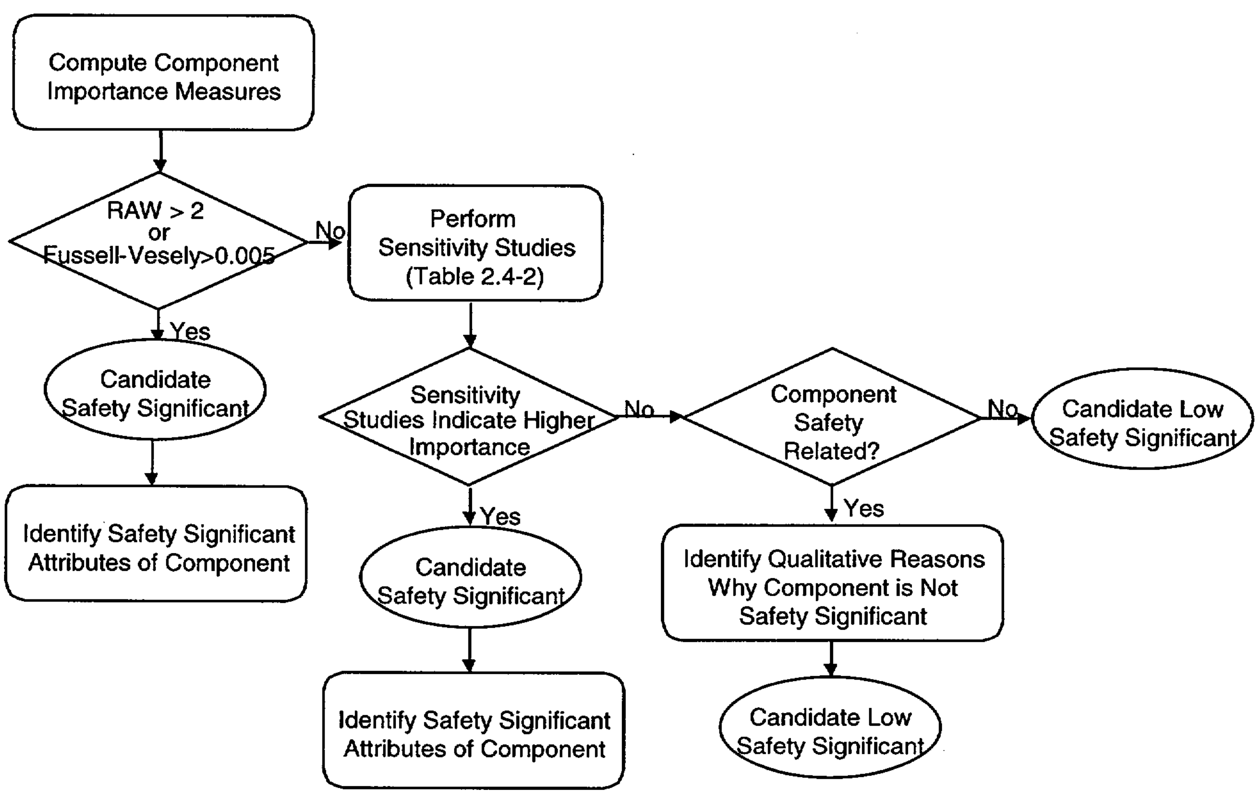
Sensitivity Study
• Increase all post accident human error probabilities by a factor of 10 or to a nominal value of 1E-3 (for small HEPs) Set all common cause failures to 0.0 Set all maintenance unavailability terms to 0.0 Increase all random failure probabilities (fail to start/open/run, etc.) by their associated error factor (e.g., 3 to 10) All manual suppression =1.0 Others?? <still under consideration>

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, it could be a candidate for RISC-3. In this case the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

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

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

Figure 2.4-4
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN FIRE PRAs



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

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

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

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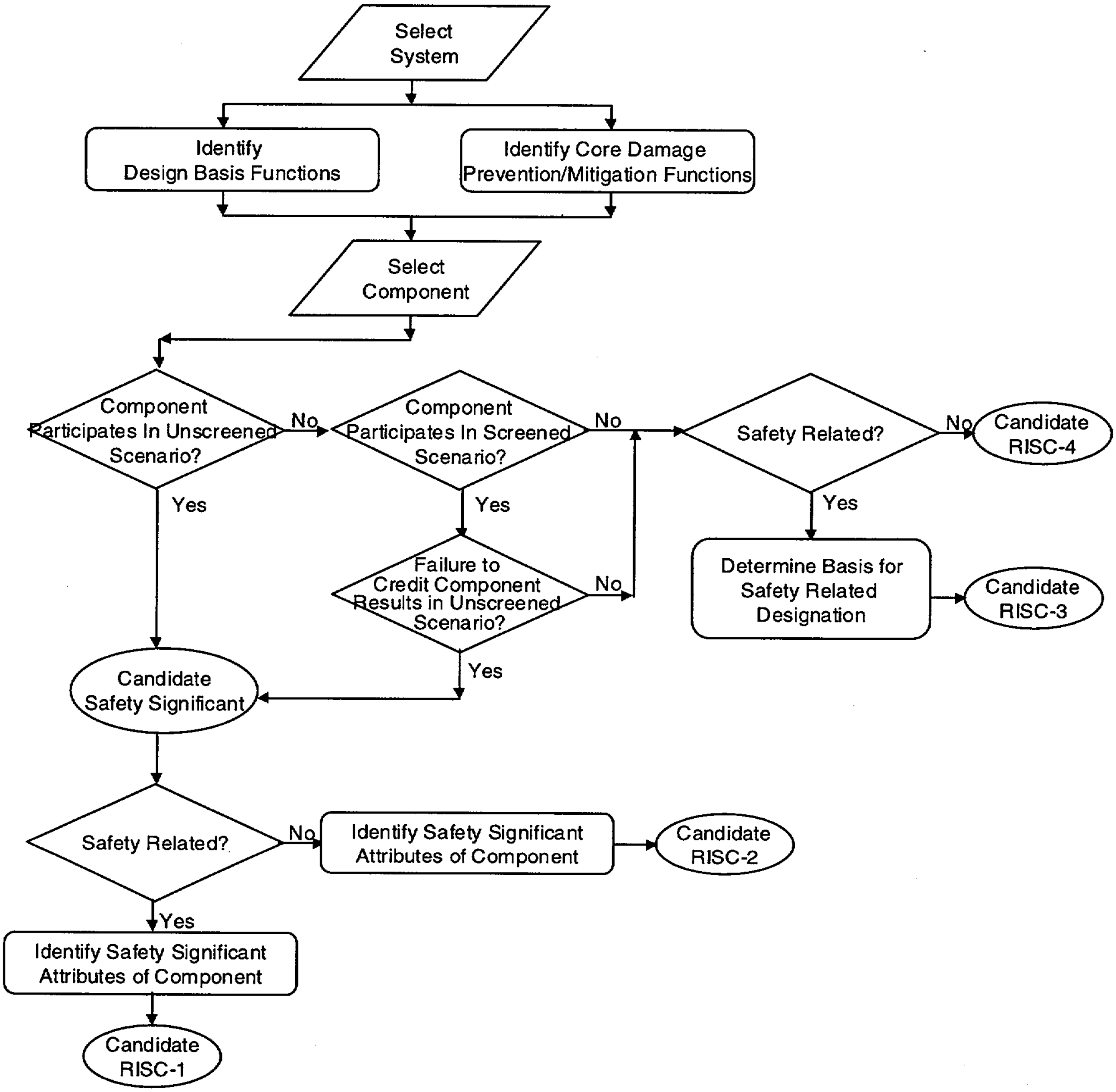
Figure 2.4-5
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN FIVE



2.4.2.3 Seismic Assessment

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

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

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

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

Table 2.4-3
Sensitivity Studies For Seismic PRA

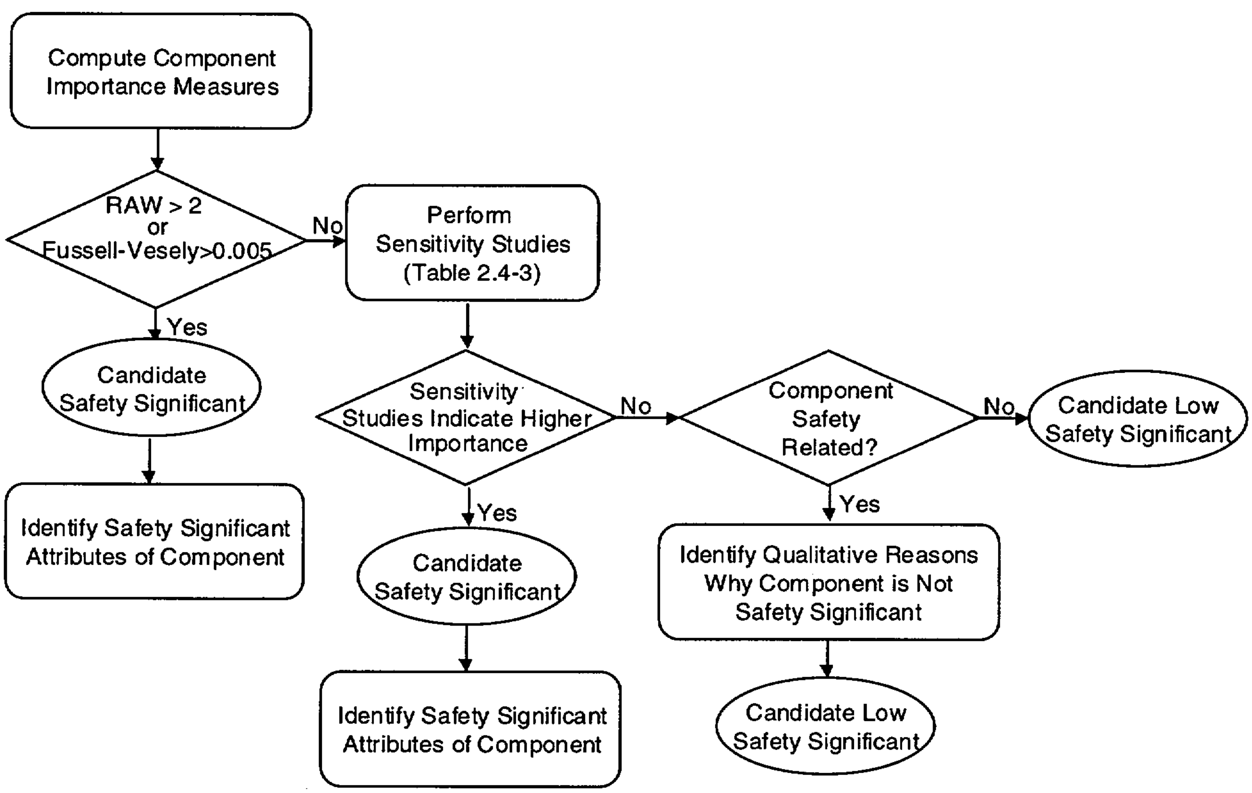
Sensitivity Study
• Increase all post accident human error probabilities by a factor of 10 or to a nominal value of 1E-3 (for small HEPs)
Set all common cause failures to 0.0
Set all maintenance unavailability terms to 0.0
Increase all random failure probabilities (fail to start/open/run, etc.) by their associated error factor (e.g., 3 to 10)
Use correlated fragilities for all SSCs in an area
Others??
<still under consideration>

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, it could be a candidate for RISC-3. In this case the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

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

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

Figure 2.4-6
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN SEISMIC PRAs



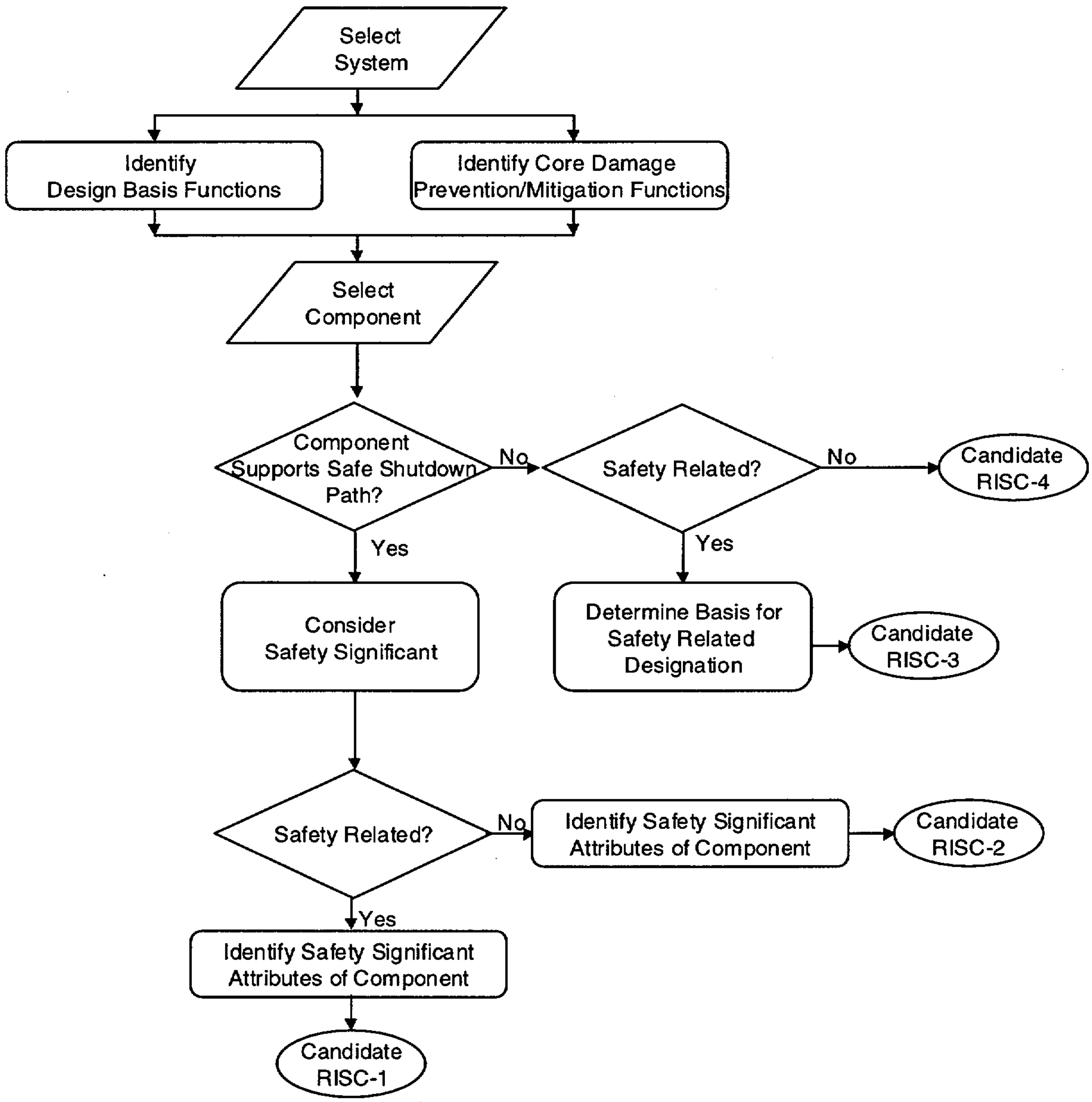
The seismic margins methodology is a screening approach to evaluating seismic hazards. It does not generate core damage values; rather, it simply assists in identifying potential seismic susceptibilities and vulnerabilities. For this reason, it is somewhat limited in being able to support the identification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

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

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

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

Figure 2.4-7
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN SEISMIC MARGINS



2.4.2.4 Other External Hazard Assessment

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

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

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

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

Table 2.4-4
Sensitivity Studies For Other External Hazard PRA

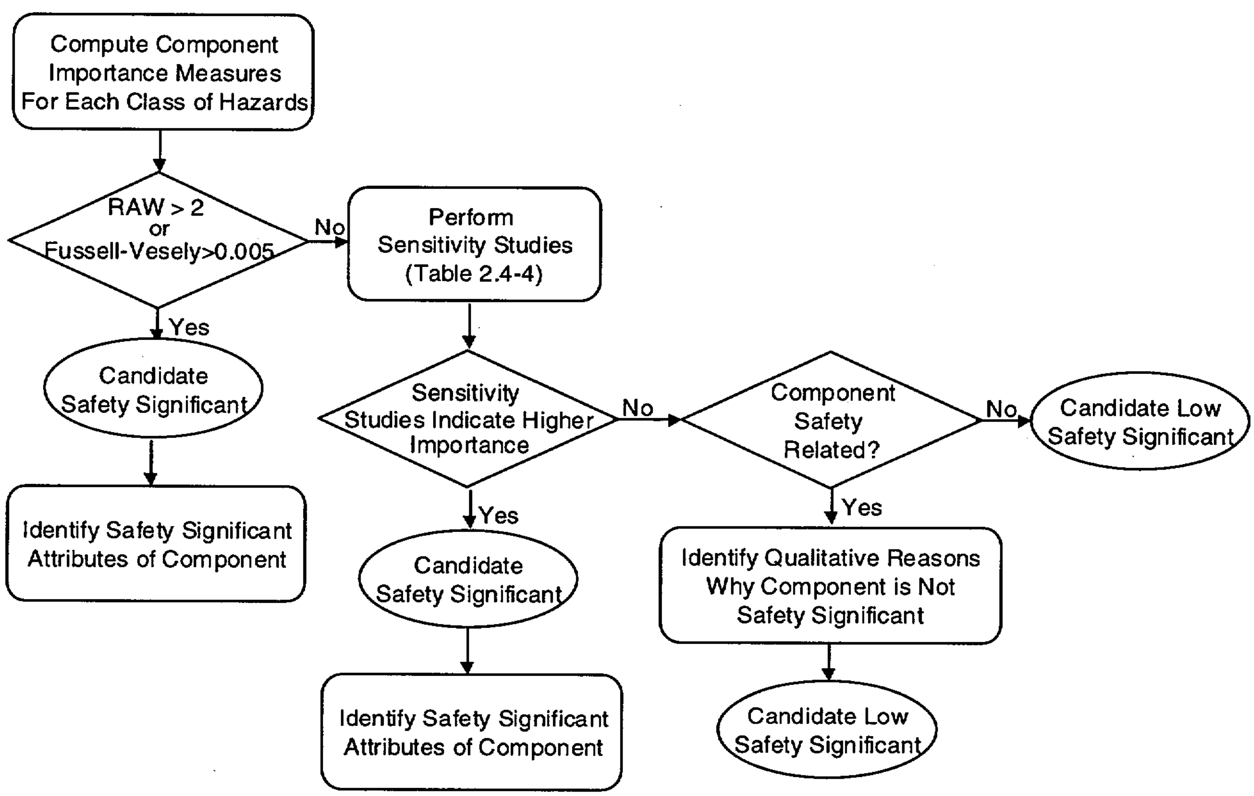
Sensitivity Study
• Increase all post accident human error probabilities by a factor of 10 or to a nominal value of 1E-3 (for small HEPs) Set all common cause failures to 0.0 • Set all maintenance unavailability terms to 0.0 Increase all random failure probabilities (fail to start/open/run, etc.) by their associated error factor (e.g., 3 to 10) Others?? <still under consideration>

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, it could be a candidate for RISC-3. In this case the analyst is expected to define why that component is of low risk significance (e.g., doesn't perform an important function, excess redundancy, low frequency of challenge, etc.).

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

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

Figure 2.4-8
RISK IMPORTANCE ASSESSMENT PROCESS FOR COMPONENTS ADDRESSED IN EXTERNAL EVENT PRAs



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

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

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

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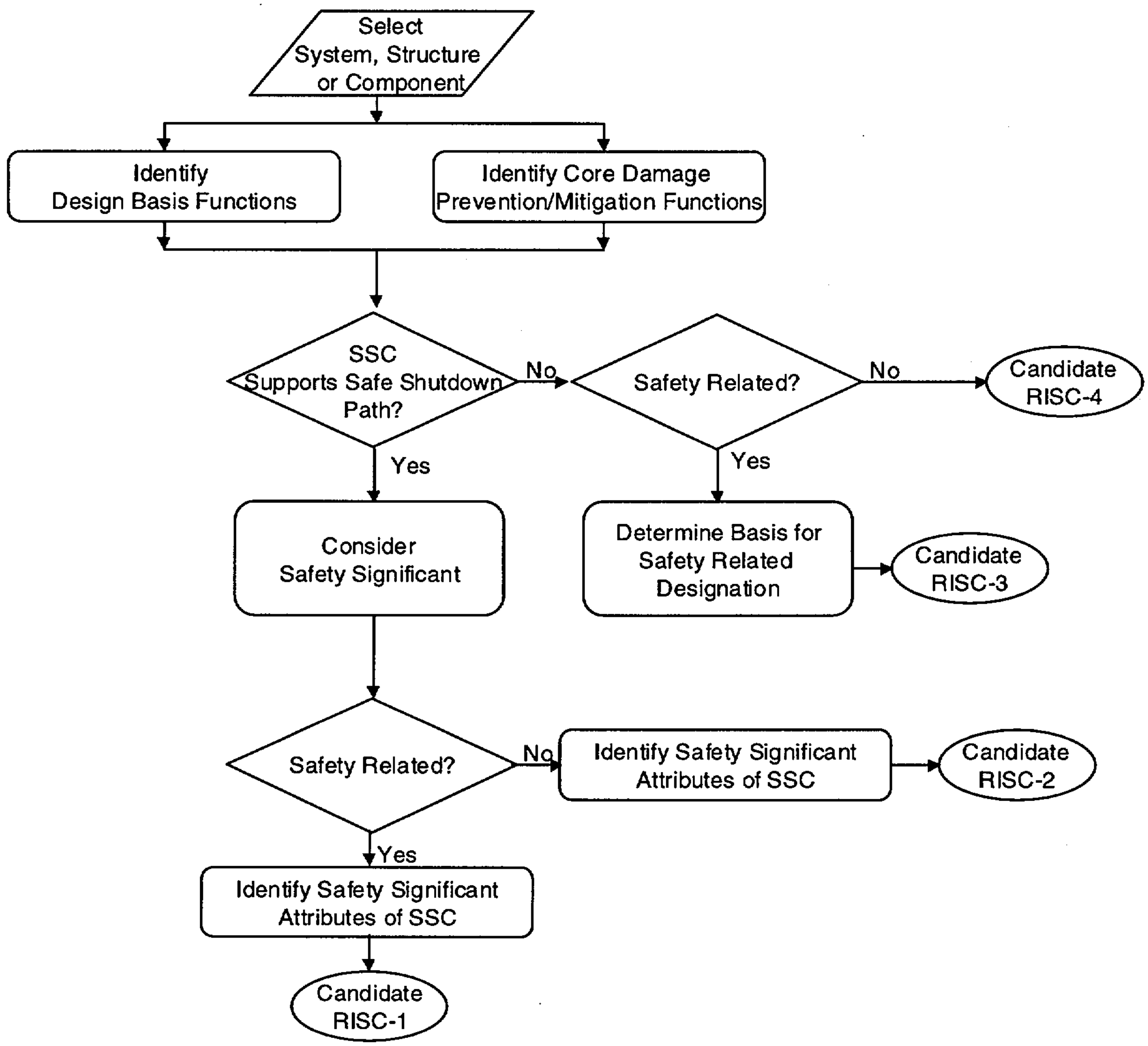
Figure 2.4-9
SAFETY SIGNIFICANCE PROCESS FOR SYSTEMS AND COMPONENTS ADDRESSED IN EXTERNAL EVENT SCREENING



2.4.2.5 Shutdown Assessment

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

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

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

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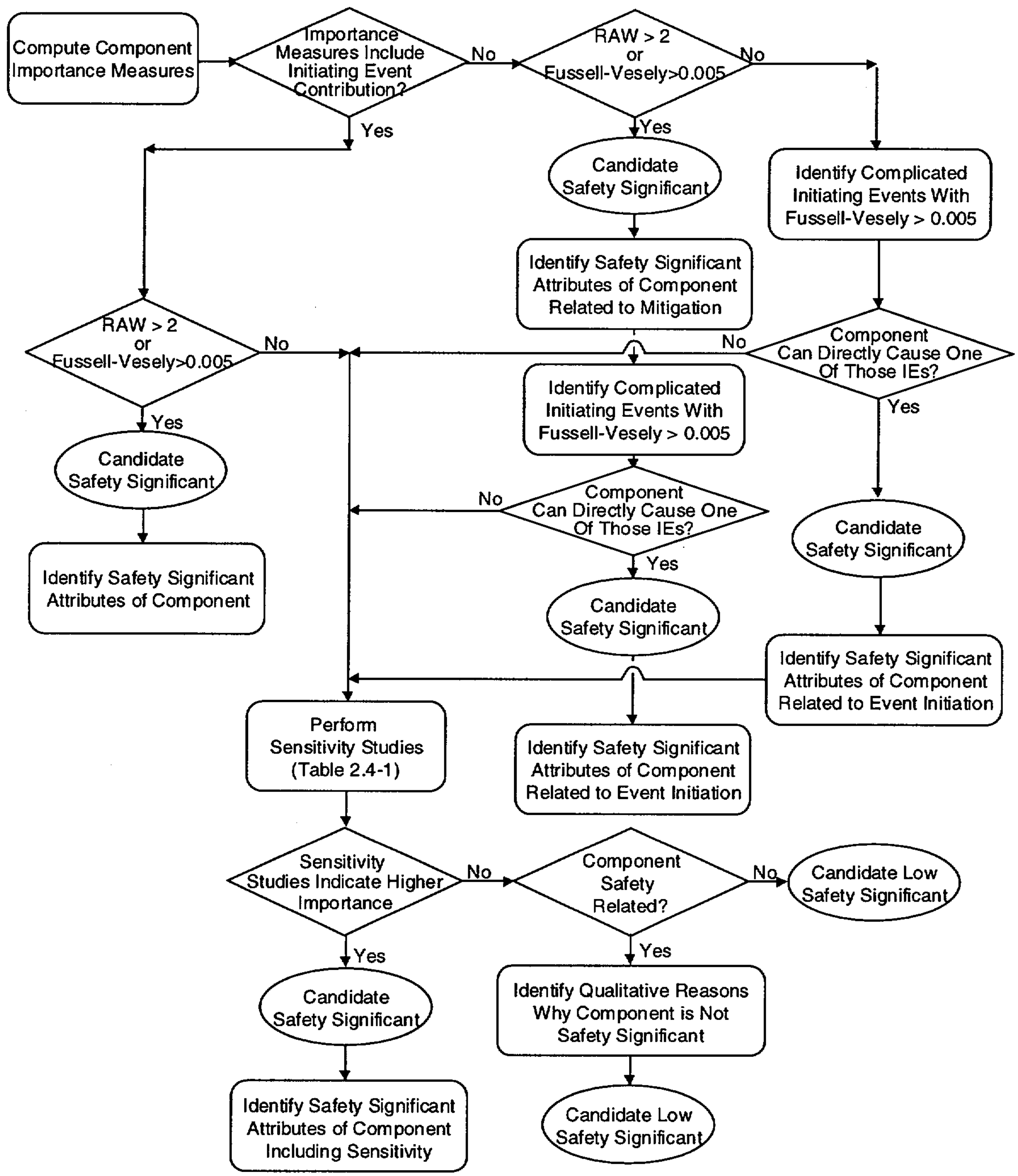
Figure 2.4-10
RISK IMPORTANCE ASSESSMENT PROCESS
FOR COMPONENTS ADDRESSED IN
LOW POWER/SHUTDOWN PRAs
(Same as Internal Event PRA)



Meeting the guidelines for shutdown safety identified in NUMARC 91-06 is not equivalent to a shutdown PRA and does not generate quantitative information comparable to core damage values. Rather it simply attempts to assure 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 2.4-11.

In this process a component can be identified as safety significant for shutdown conditions for one of two reasons: (1) it is identified in the licensees NUMARC 91-06 program implementing procedures as a primary shutdown safety system or (2) it could initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.) If the component does not participate in either of these manners, then it is considered candidate low safety significant with respect to shutdown safety.

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

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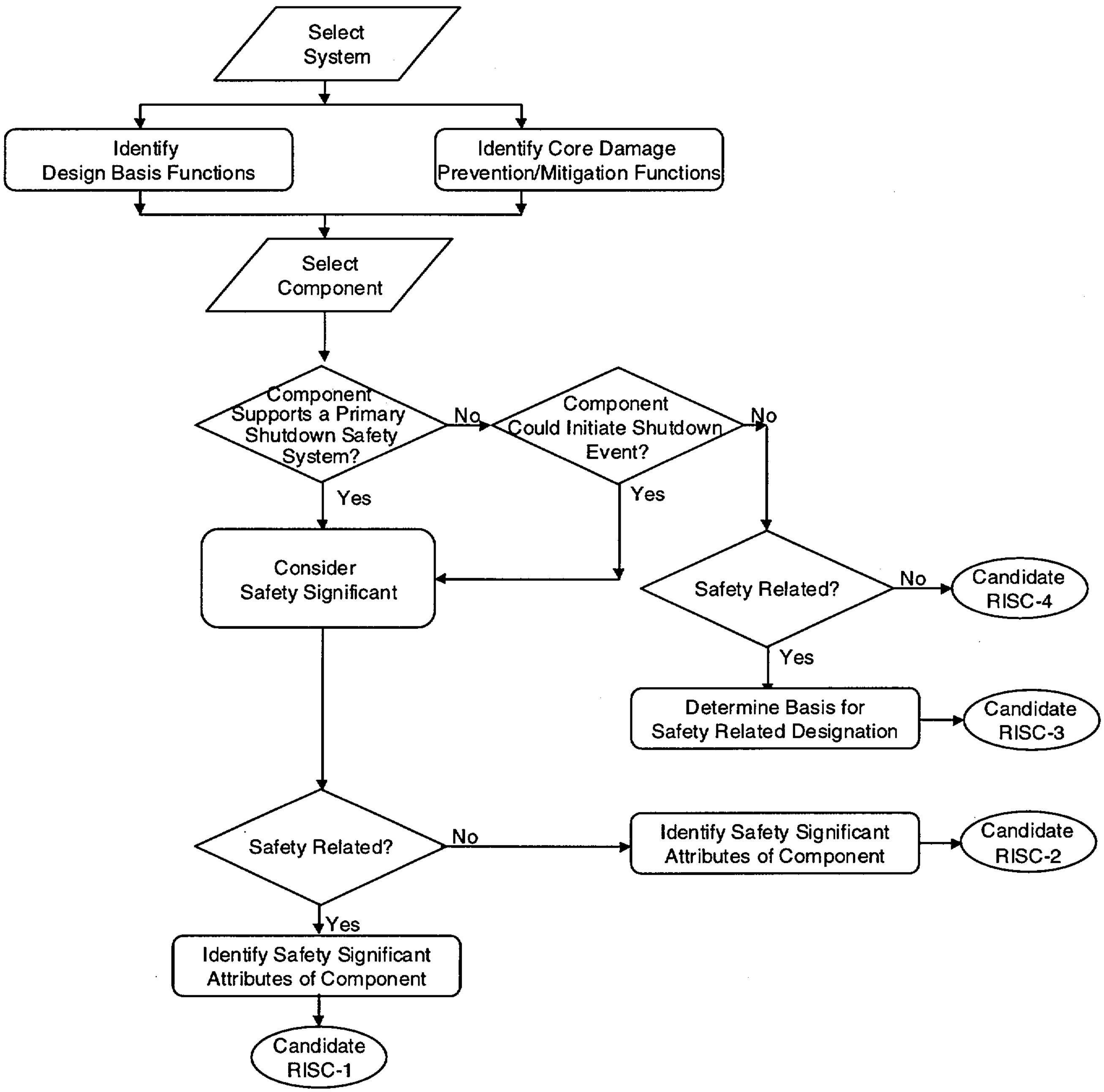
Figure 2.4-11
SAFETY SIGNIFICANCE PROCESS FOR
SYSTEMS AND COMPONENTS CREDITED IN NUMARC 91-06 PROGRAM



2.4.2.6 Integral Assessment

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

Integrated Fussell-Vesely Importance

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

Where,

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

Integrated Risk Achievement Worth Importance

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

Where,

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

Integrated Risk Reduction Worth Importance

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

Where,

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

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

2.5.2.7 Presentation of Risk Information

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

DRAFT

Figure 2.4-12

EXAMPLE RISK-INFORMED SSC ASSESSMENT WORKSHEET

SSC(s) Evaluated: ______________________________

Safety Related: Yes { No {

Design Basis Function(s) Supported: ______________________________

PRA Functions Supported: ______________________________

____

____

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

Defense In Depth/Common Cause Assessment:

____

____

____

Basis for SSC Classification: ______________________________

____

____

Safety Significant Attributes: ______________________________

____

____

2.4.3 Integrated Decision-making Panel Review & Classification

UNDER DEVELOPMENT

2.4.4 Evaluation of Recommended Changes

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

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

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

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

The second step is to perform sensitivity studies using the available PRAs to evaluate the potential impact on CDF and LERF. This step is useful because the importance measures used in the initial safety significance assessment were based on the individual SSCs considered. Changes in performance can influence not only the importance measures for the SSCs which have changes in performance, but also others. Thus, the aggregate impact of the changes should be evaluated to assess whether new risk insights are revealed. Sensitivity studies should be realistic. For example, increasing the unreliability of RISC-3 SSCs by a factor of 2 to 5 could represent a bounding impact on SSC performance. Likewise, reducing the unreliability of RISC-1 and RISC-2 SSCs by a similar factor may be called for, depending upon the specific changes in special treatment. The changes in CDF and LERF computed in such sensitivity studies should be compared to the risk acceptance guidelines of Reg. Guide 1.174 as a measure of their acceptability. In addition, importance measures from these sensitivity studies can provide insight as to which SSCs and which failure modes are most significant.

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

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

Section 3

treatment of risk-informed safety class structures, systems and components

UNDER DEVELOPMENT

Section 4

DOCUMENTATION & APPROVAL

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

The following shall be documented and available for NRC review:

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

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

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

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

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

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

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

NRC Review And Approval

As per 10 CFR 50.69, a licensee wishing to adopt a risk-informed SSC scope will notify the Commission in writing of its intent to implement this voluntary option. The notification letter will list:

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

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

Periodic Review

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

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

- Review of changes to plant activities that could impact the categorization results.
- Review of plant specific operating experience and data that could impact the categorization results.
- Assessment of the impact of the three elements listed above on the risk-informed SSC categorization by the Integrated Decision Process Panel
- Recommendations to change categorization.

section 5

Change control process for risk-informed sscs and activities

UNDER DEVELOPMENT

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Attachment to Commercial Quality Attributes Paper

Typical Element Examples in a Commercial Work Control Program Applicable to RISC-2 and RISC-3

Introduction

The current effort to risk-inform regulations includes the categorization of systems, structures and components (SSCs) into one of four categories based upon their safety significance. Special treatment requirements would be applied to each SSC based upon its assignment to one of the four "RICS classes." The SSC categories are:

- RISC-1 SSCs are safety-related that have been determined to be safety significant by a risk-informed categorization process.
- RISC-2 SSCs are nonsafety-related, but have been determined to be safety significant by a risk-informed categorization process.
- RISC-3 SSCs are safety-related, but have been determined to be of low safety significance by a risk-informed categorization process.
- RISC-4 SSCs are nonsafety-related and are of low safety significance, and are consequently not subject to NRC regulations.

SSCs that are classified as RISC-2 and RISC 3 require appropriate treatment to assure that they will perform their required function(s) with reasonable assurance and reliability, yet to a lesser degree than safety-related/safety significant SSCs (RISC-1 SSCs). Under 10 CFR 50.65, nonsafety-related SSCs are included in the regulatory scope in regard to monitoring, goal setting and corrective action. These nonsafety-related SSCs, which are subject to commercial level programs and procedures, have satisfied the §50.65 performance criteria. As such, these commercial programs are proven and appropriate levels of controls for RISC-2 and RISC-3 SSCs.

Many plants do not have a specific program labeled "commercial quality program." Rather, the elements and procedures are a compendium of numerous plant specific programs and procedures, that when combined together, provide reasonable assurance that the functions (power production and safety) of these nonsafety-related SSCs will be satisfied. In specific instances, such programs and procedures may be a subset of the more formal 10 CFR 50, Appendix B quality programs and procedures. These commercial programs are in place, have been, and continue to be

DRAFT

the vehicle for providing reasonable assurance that the functions associated with the §50.65 performance criteria are satisfied.

These general quality program elements for balance-of-plant (power generation) activities are disseminated throughout plant procedures. The following elements are examples of the type of programs and procedures that are included a licensee's work control program for balance-of-plant equipment.

I. Maintenance Rule Program

Implements the Maintenance Rule at the station, including SSC scoping and monitoring, classifying SSC performance in accordance with criteria and goals, ensuring proper corrective actions when performance criteria are not met, and periodically evaluating overall program performance. [Neither MR goal setting (a)(1) nor risk assessment (a)(4) is considered necessary for the RISC 3 classifications due to their low safety significance, and could be eliminated in a "graded" approach to using the MR 3 systems].

II. Corrective Action Program

Establishes the measures to be taken to assure the conditions adverse to quality (e.g., failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances) are promptly corrected. For conditions adverse to quality, this program establishes measures to provide reasonable assurance that the cause of the condition is determined, and that corrective action is taken in a timely and accurate manner, consistent with safety significance and power production requirements. The program provides for the resolution of overdue corrective action through escalation to higher levels of management, and for the trending of deviating conditions.

III Preventive Maintenance (PM) Program

Establishes the requirements and guidelines for the development and implementation of preventive maintenance to ensure plant equipment is maintained at a quality level to perform its intended function. The program includes the identification, maintenance, and scheduling of PM on permanent installed plant equipment, equipment in storage, and maintenance equipment.

IV Design Change Program

Establishes the process for managing the preparation, implementation, and where necessary, the licensing of design changes to systems, structures and components (SSCs). It defines the controls necessary to ensure safe implementation of station

DRAFT

design changes. As necessary and appropriate, post-modification testing is performed to determine or verify the capability of a modified SSC to meet specified design requirements and design bases.

V Procurement Program

Procurement of SSCs is controlled by administrative procedures that implement quality assurance program elements for procurement and materials management consistent with safety and power generation. As necessary, and consistent with the safety-significance or power production requirements, the program includes: vendor surveillance audits and maintenance of approved vendor lists, receipt inspection, materials verification activities, and special handling and storage procedures.

VI Procedure Program

This program applies to technical and administrative procedures and includes the necessary processes to maintain procedure quality. The program further establishes the processes for 1) the development, review, and approval of new procedures, procedure revisions, procedure changes and procedure deletions, 2) review and approval of vendor procedures, and 3) performance of periodic procedure reviews.

VII PRA Update Program

This program incorporates a feedback process to update the PRA on frequency determined by circumstances, but not to exceed 36 months. Where appropriate and necessary, the update incorporates applicable “state-of-the-art” changes relative to PRA technology, as well as plant design changes that would affect the PRA and have been implemented since the last update.

VIII Work Control

This program provides the process for identifying, controlling, and documenting work activities, including implementing design changes, at the station. The program ensures that the processing of work requests and work order tasks supports the completion of work in a safe, timely and efficient manner such that safe and reliable plant operation is optimized.

IX Work Planning and Scheduling

This program provides the requirements and guidelines for planning and scheduling maintenance and other work activities at the Station to maximize plant operational

DRAFT

safety, reliability and availability. The program addresses the planning and scheduling of the following activities:

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

On-line maintenance

Periodic testing

Installation of design change packages

X Other Programs

Other programs that also promote reasonable assurance and reliability that specific RISC-2 and 3 SSCs will adequately perform their function(s) include:

Secondary Piping Inspection and Replacement Program (includes condensate and feedwater systems, etc. subject to flow-assisted corrosion)

Diesel Generator Reliability Program (includes AAC diesel)

Technical Requirements Manual (Fire Protection/Appendix R equipment, AAC equipment, snubbers, etc.)

COMMENTS FROM ASME BNC&S TG ON RIP50
by
C. Wesley Rowley (Chairman of TG)

(1) Need to return to the fundamental principles--the risk ranking is done to focus on the most significant items--to now put emphasis on items that are not in this category defocuses the spotlight on reactor safety and achieves nothing in real terms. The choice is use the best information available and move into the future or cling to the past. If the components are truly risk ranked, then this issue goes away --those items important to safety still receive a full treatment; the others, appropriately, receive commercial treatment

(2) If the term "commercial practice" is balance-of-plant Codes & Standards, then it is primarily Section VIII and B31 Pipe Codes. The Board on Pressure Technology has the Post Construction Committee (PCC) developing a Risk Based Inspection Planning Standard that is in final draft; they are also adapting API 579 for fitness-for-service evaluations of service induced degradation, and codifying special repair procedures. An "updated Section XI" is emerging in this PCC, but with a much broader scope, that is addressing plant aging for a multitude of industries.

(3) Special Treatment for Repair / Replacement / Modification (some thoughts):

(3a) In IWA-4131.2, we provide reduced requirements for small items that were previously exempt from Section XI under the old IWA-7400 1" and under exemption. In the action that eliminated the exemption and incorporated these reduced requirements, we noted that we wanted the items to perform their design function. But since they had been previously exempt, we did not need to apply a lot of the current Section XI requirements which were not felt necessary to assure the items would perform as designed. Since the items still were safety related, we did not eliminate the requirement for using a QA Program. Under the elimination of special treatment requirements, the industry wants to delete QA Program requirements, which may be all right, so that even the QA Program requirement in IWA-4131.2 could possibly be deleted. Other requirements already eliminated in IWA-4131.2 include NCA-3800, use of a Repair/Replacement Plan, possession of a Section III Certificate of Authorization by the component manufacturer, agreement with an AIA, and completion of the Section XI NIS-2 data report for the Repair/Replacement Activity.

(3b) It seems that a key requirement to have the item remain functional, is to meet its design requirements. Therefore, these RICS-3 items should continue to meet the materials, design, fabrication and examination requirements of their original Construction Code, without QA Program requirements and administrative requirements (NCA-3800 and NCA-4000 in Section III). Section XI could have a Mandatory Appendix to identify how to apply Section XI repair / replacement activity provisions to RISC-3 items and would identify the limited repair/replacement activity requirements to apply. The Appendix would draw from experience in the non-nuclear B&PV industry and identify what we thought was necessary for commercial practice. For replacing components, it might allow going to a non-nuclear Construction Code, but for repair and replacement of materials and parts in an existing component, Owners would want to continue with their existing design with reduction in the QA and administrative requirements needed to be met. This could get very detailed but may not be acceptable to the nuclear industry. Therefore, we could make it provide the general requirements. A key to how to write such an Appendix would be to know how the NRC would use it!

(4) Other Special Treatment Thoughts: How should we handle the need for seismic and environmental qualification of commercial grade (commercially accepted practice) components? When the Committees are working on this, they need to pay careful attention to 10CFR21 requirements relating to the dedication of commercial grade items to safety related service.

(5) Overall Perspective (perhaps the tie between Option #2 and Option #3): as an industry we have not fully linked design basis, PRA, maintenance rule, and license renewal. As a result we collect some design basis information that, while interesting, adds nothing to reactor safety, we do PRA primarily at the system level so that the application of the rules to components that are not important to safety continues, we have not risk ranked the maintenance rule application so we do some things that really add nothing to reactor safety, and finally the license renewal process is pointed toward component aging with little emphasis on importance to reactor safety. There are huge gains to be made in reactor safety AND plant economics by an approach that recognizes where we are headed as an industry as opposed to the separate little kingdoms we are currently addressing.

30 March 2000
RIP 50 Meeting
at White Flint