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SOUTHERN COMPANY

May 17, 2013

Docket Nos.: 50-424
50-425

NL-13-0997

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Response to Request for Additional Information

Ladies and Gentlemen:

By letter dated August 31, 2012, Southern Nuclear Operating Company (SNC) requested amendments to the Vogtle Electric Generating Plant (VEGP) (TAC NOS ME9472 and ME9473). The proposed amendments would revise the VEGP licensing basis to implement 10 CFR 50.69, risk informed categorization and treatment of structures, systems, and components for nuclear power plants.

By letter dated April 17, 2013, the NRC requested additional information. The enclosure provides the response to the NRC's request for additional information. As discussed during a conference call, between SNC and NRC staff, held on May 1, 2013, the responses to RAIs 19, 25, 26, and 27 will require additional time to develop and will be provided within 120 days of the date of this letter.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. C. R. Pierce states he is Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

Sworn to and subscribed before me this 17th *day of* May, *2013.*

Nancy Louise Henderson
Notary Public

My commission expires: March 23, 2014

Enclosures: 1. Response to Request for Additional Information Regarding Pilot 10 CFR 50.69 License Amendment Request

2. Requested Procedures

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. E. Tynan, Vice President – Vogtle
Mr. B. L. Ivey, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Fleet Operations
RType: CVC7000

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Senior Project Manager - Vogtle
Mr. L. M. Cain, Senior Resident Inspector – Vogtle

State of Georgia
Mr. J. H. Turner, Environmental Director Protection Division

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Response to Request for Additional Information

Enclosure 1

Response to Request for Additional Information
Regarding Pilot 10 CFR 50.69 License Amendment Request

NRC RAI #1

SNC provided the Nuclear Regulatory Commission (NRC) staff a set of draft procedures under cover letter dated August 17, 2011 (Reference 1). Please confirm that these are the current procedures or provide the latest versions of these procedures.

SNC Response

The latest version of 10 CFR 50.69 categorization related procedures are attached to this letter (Enclosure 2). The enclosure contains the following procedures:

Number	Title	Version
NMP-ES-065	10 CFR 50.69 Program	Version 1.0
NMP-ES-065-001	10 CFR 50.69 Active Component Risk Significance Insights	Version 1.0
NMP-ES-065-002	10 CFR 50.69 Passive Component Categorization	Version 1.0
NMP-ES-065-003	10 CFR 50.69 Risk Informed Categorization for Structures, Systems, and Components	Version 1.0
NMP-ES-066	General Guidance for Decision-Making Panels – 50.69	Version 2.0
NMP-ES-066-002	Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities	Version 1.0
NMP-ES-066-002-F01	Risk Informed Categorization Integrated Decision Making Panel Qualification Form - 50.69	Version 1.0
NMP-ES-066-002-F02	Risk Informed Categorization Integrated Decision Making Panel Meeting Minute Form -50.69	Version 1.0

While applying the above procedures to trial categorization of three systems, the following four changes were identified. These changes will be incorporated during the next revision, which will happen after receiving Safety Evaluation Report (SER) from the NRC for the Vogtle Electric Generating Plant Pilot 10 CFR 50.69 License Amendment Request submitted to NRC on August 31, 2012 (ML12248A035).

Change #1:

Move section 5.22 (Risk Sensitivity Study Background) into the desktop guideline.

Change #2:

In sections 5.23 (Perform Initial Sensitivity Study) and 5.24, of NMP-ES-065-001, revise to mention increasing unavailability (UA) by the same factor for a Low Safety Significance (LSS) component(s) that has unavailability basic event. The current version of NMP-ES-065-001 (Version 1) does not increase UA.

Refer to NEI page 59 "...The risk sensitivity study should be performed by manipulating the unavailability terms for PRA basic events that correspond to components that were identified in the categorization process as having low safety significance because they do not support a safety-significant function..."

Change #3:

Revise, section 5.23 and 5.24 of NMP-ES-065-001 to clearly state that failure rates of the basic and common cause events MUST be raised by a chosen factor. For the trial categorization, a factor of 3 has been selected. Therefore, make the following changes:

Delete 5.23.3 in the current version (which is 1.0). Then make the following changes for 5.23 and 5.24:

5.23 Perform Initial Sensitivity Study

5.23.1 No change

5.23.2 Perform this sensitivity study for the system that is being categorized and provide results to the IDP.

5.23.2.1 A factor of 3 has been selected for initial and cumulative sensitivity study (per 5.24).

5.23.2.2 Increase unreliabilities of ALL candidate LSS SSCs modeled in the PRA by a factor of 3.

5.23.2.3 Increase unavailability by a factor of 3 for those candidate LSS SSCs whose unavailabilites have been modeled in the PRA.

5.23.2.4 Same wordings as 5.23.4 in the current version.

5.23.2.5 Determine if the quantitative acceptance guidelines outlined in the Regulatory Guide 1.174 have NOT been exceeded.

5.23.3 Same wordings as 5.23.5 in the current version.

5.24 Perform a cumulative sensitivity study for ALL LSS components modeled in the PRA for ALL systems that have been categorized and the system that is being categorized by repeating steps 5.23.2.1 through 5.23.3.

Change #4:

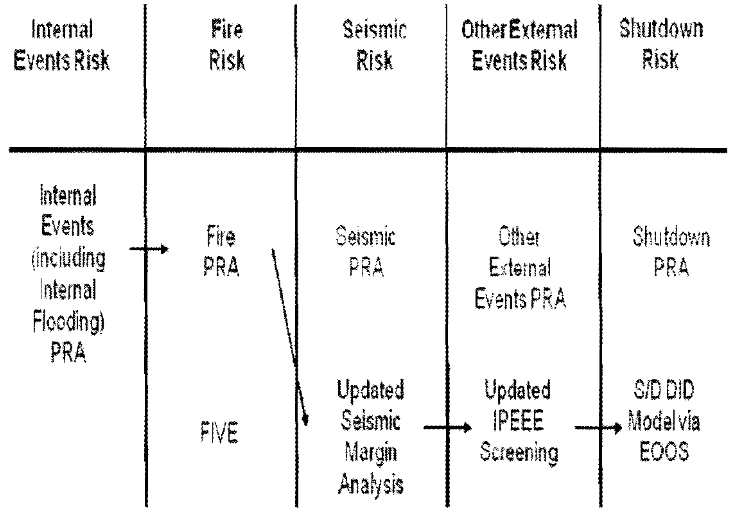
NMP-ES-065-003 will be revised to ensure that when EOPA or EOPC answer is "No", the alternative means are proceduralized and included in Licensed Operator training.

NRC RAI #2

Nuclear Energy Institute (NEI) 00-02 (Reference 2) provides alternative acceptable methods for several categorization tasks. For example, a peer reviewed fire probabilistic risk assessment (PRA) could be used as is internal events PRA, or all unscreened structures, systems, and components (SSCs) in a Fire-Induced Vulnerability Evaluation (FIVE) analysis could be assigned as High Safety Significant (HSS). The draft procedures also include alternative methods to perform individual tasks. When alternative methods are included in the procedures, how is one of the methods selected for use? Under what conditions can different methods be used for different systems?

SNC Response

In the Plant Vogtle Electric Generating Plant Pilot 10 CFR 50.69 License Amendment Request submitted to NRC on August 31, 2012, (ML12248A035), SNC is requesting NRC to approve use of Probabilistic Risk Analysis (PRA) to assess the following two hazards – Internal Events (including Internal Flooding) and Fire. After a Safety Evaluation Report (SER) is obtained from NRC for the aforementioned LAR, the procedure will be revised to indicate that Internal Events (including internal Flooding) and Fire PRAs shall be used to assess hazards when categorizing Structures, Systems, and Components (SSCs) at Plant Vogtle. Fire-Induced Vulnerability Evaluation (FIVE) analysis shall not be used. The following pictorial aid summarizes how each hazard will be assessed when categorizing SSCs at Plant Vogtle. This will be made clear in the procedure after a SER is received. This will be consistent with Enclosure 3, Operating Licenses Clean Typed Pages, of the LAR (ML12248A035) that has proposed a license condition that states, "NRC prior approval is required for a change to a categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment)."



NRC RAI #3

Section 3.3.1.2 of Enclosure 1 to the license amendment request (LAR) in Reference 3, states that, "In May 2009, the VEGP PRA internal events model Revision 4 (including internal flooding) was reviewed against the requirements of [ASME RA-Sc-2007, Reference 4] as amended by RG 1.200, Revision 1...[Reference 5]"

Please summarize the peer review conducted in May 2009 and clarify if it was a full peer review where the team met the guidelines outlined in NEI 00-02 (e.g., 5 or 6 members that included the full range of experience required to perform an internal events PRA), followed the process outlined in NEI 00-02 (e.g., offsite preparation, one week onsite review, and post review documentation), and reviewed the PRA against all the elements in the ASME 2009 standard. If the review was not a full peer review, please describe the review in detail and provide all earlier Findings and Observations (F&Os) from any previous reviews.

SNC Response

Clarification:

Although there is no technical impact, SNC provides the following clarification. Plant Vogtle License Amendment Request (LAR) made a reference to ASME RA-Sc-2007, which is not technically correct because the May 2009 peer review report references RA-Sb-2005. Both RA-Sb-2005 and RA-Sc-2007 are Addenda to ASME PRA Standard RA-S-2002. Addendum c of RA-S-2002 made only relatively minor changes to Addendum b, and these changes do not have any technical impact on the capability of the PRA. The main changes of interest between these two addenda are in Section 5 (Configuration Control), particularly sections 5.5 (Pending Changes) and 5.6 (Previous PRA Applications). In the RA-Sc-2007 addenda, additional verbiage was incorporated to provide further clarifications to a user in these two sections. Hence, the changes made in these two sections were clarification type changes. The RA-Sc-2007 addenda also added non-mandatory Appendix A to provide examples of PRA Maintenance, PRA Upgrade, and the Advisability of Peer Review, and made editorial corrections to several references in Section 4.

In February 2009, the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) approved the new combined PRA Standard (ASME/ANS RA-Sa-2009, "Addendum to ASME/ANS RA-S-2008 – Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", the American Society of Mechanical Engineers and the American Nuclear Society, February, 2009). NRC endorsed this new standard in Revision 2 of RG 1.200, which was issued in March 2009. The new standard was not generally available in time to support Vogtle peer review, so the peer review was performed against the version of the ASME PRA Standard (RA-Sb-2005) that had been used in the PWR Owners Group peer reviews of internal events at power PRAs up to that point. For this reason, Table 6 in Enclosure 1 of Vogtle 50.69 LAR compares the 2007 version (RA-Sc-2007) against the 2009 version (RA-Sa-2009). The peer review concludes that the Vogtle model satisfies the guidance of RG 1.200, Revision 2. The most significant change in the PRA Standard between RA-S-2002 and addenda and RA-S-2008 and addenda was the addition of requirements for PRAs for other than internal events at power. That is, the requirements for internal events at power PRAs in RA-Sb-2009 are substantially the same as those in RA-Sb-2005 (and RA-Sc-2007).

<u>Summary of May 2009 Peer Review</u>:

The scope of the peer review conducted in May 2009 was a full scope PRA peer review of the Plant Vogtle internal events at power PRA to determine compliance with ASME PRA Standard (RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, December 2005) and RG 1.200, Revision 1. This peer review was performed using the process defined in Nuclear Energy Institute (NEI) 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard". NEI 05-04 has been endorsed by Revision 1 and Revision 2 of RG 1.200. Note that, although the title of NEI-05-04 includes the phrase "Follow-on", this document provides a process appropriate for both full and partial peer reviews of internal events at power PRAs against the requirements in the PRA Standard and RG 1.200.

The peer review was conducted during the week of May 4 through May 8, 2009. It covered all nine technical elements from the ASME PRA Standard plus the configuration control element. The model that was reviewed was the "VEGP Level 1 and Level 2 PRA Model Revision 4 - at power, internal events".

The peer review team consisted of six (6) reviewers having a full range of experience required to perform the peer review. Each reviewer was assigned a lead role for a review element. The lead reviewer was assisted by two reviewers acting in a support role. The documents were supplied in advance to the peer review team members. During the week of May 4 through May 8, 2009, the peer review team was onsite, performed the review, and provided preliminary results on May 8, 2009. A final report was issued on November 10, 2009. The following table summarizes results of the peer review.

Summary of Capability Category Assessment by PRA Element									
	Capability Category								
SR	Not Met	Met	CC-I	CC-II	CC-III	CC-I/II	CC-II/III	N/A	TOTAL
Initiating Event (IE) Total		21	0	5	0	5	0	2	33
Accident Sequence Analysis (AS) Total		17	0	1	2	0	0	1	21
Success Criteria (SC) Total		10	0	1	0	0	3	0	14
Systems Analysis (SY) Total		32	0	2	0	2	3	3	42
Human Reliability (HR) Total	1	19	0	5	1	2	6	1	35
Data Analysis (DA) Total		17	0	5	2	2	4	4	34
Internal Flooding (IF) Total		39	0	2	1	3	2	3	50
Quantification (QU) Total	1	28	0	2	1	0	2	1	35
Large Early Release Frequency (LE) Total	1	17	0	15	0	0	4	5	42
Maintenance &Update/Configuration Control (MU) Total		10	0		0	0	0	0	10
GRAND TOTALS	3	210	0	38	7	14	24	20	316

NRC RAI #4

Section 3.3.2.2 of Enclosure 1 to the LAR, states that, "a focused scope peer review was conducted for the Qualitative Screening, and Quantitative Screening elements that were marked as Not Reviewed by the [fire PRA] peer review team." Please summarize this focused scope review and compare it with the [focused scope] peer review guidance described in ASME/ANS RA-S1-2009, Section 1-6.2.4(d).9 (Reference 6).

SNC Response

During the week of February 13, 2012, a full scope peer review was conducted for the Vogtle fire PRA. Although the peer review was intended to be a full scope peer review, the peer reviewers informed SNC in the exit meeting that two technical elements, Qualitative Screening (QLS) and Quantitative Screening (QNS), were not included in the review. Therefore, a focused peer review was performed by two industry experts - Jim Chapman and Paul Amico, who have many years of experience in PRA and fire PRA. The purpose of the Focused Peer Review was to review QLS and QNS elements, which were not reviewed during the peer review in February 2012.

The Focused Peer Review of the Vogtle Fire Probabilistic Risk Assessment (FPRA) was performed per the requirements outlined in Part 4 of the ASME/ANS PRA Standard (ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, 2009). This peer review was performed using the process in Nuclear Energy Institute (NEI) 07-12 (NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Nuclear Energy Institute, June 2010). During the focused peer review, in addition to reviewing QLS and QNS technical elements, the resolutions of F&Os for the technical element Plant Partitioning (PP) were also reviewed for the following two reasons - the qualitative screening conducted for Vogtle was included in the calculation, which addresses both PP and the QLS element used input from the PP element. The following table provides a description of the elements and the scope of the focused peer review. A peer review team consisted of two reviewers who reviewed technical elements during the onsite review session on May 10, 2012.

Element ID	Description	In Scope of Focused Peer Review
PP	Plant Partitioning	No, but considered F&Os and their resolutions
ES	Equipment Selection and Location	No
CS	Cable Selection and Location	No
QLS	Qualitative Screening	Yes
PRM	Plant Response Model	No
FSS	Fire Scenario Selection and Analysis	No
IGN	Ignition Frequency	No
QNS	Quantitative Screening	Yes. Confirmed that Quantitative Screening was not performed
CF	Circuit Failure Analysis	No
HRA	Human Reliability Analysis	No
FQ	Fire Risk Quantification	No
SF	Seismic Fire Interactions	No
UNC	Uncertainty Analysis	No
MU	Maintenance and Update	No

The following table summarizes the overall results of the Vogtle Fire PRA focused peer review.

Summary of Overall Results of the Vogtle Fire PRA Focused Peer Review										
Fire PRA Element	Number of Supporting Requirements Meeting Each Capability Category									
	Not Met	Met	CC-I	CC-I/II	CC-II	CC-II/III	CC-III	Not Applicable (NA)	Not Reviewed (NR)	Total
QLS		7								7
QNS*								6		6

*Vogtle did not perform quantitative screening so the requirements of QNS are not applicable.

The focused peer review team concluded that the methodologies being used were appropriate and sufficient to meet the Standard. For the fire PRA element QLS, the review team concluded that QLS was performed using appropriate criteria, and the criteria were applied in a proper fashion. No criteria were proposed that went beyond those suggested in the Standard. For the fire PRA element QNS, the review team concluded that QNS was not performed. The peer review team verified that numerical results were generated and retained in the quantification results for all Physical Analysis Units (PAUs) that were not qualitatively screened. Therefore, all QNS SRs are N/A.

There were three suggestion level F&Os for technical element QLS. These suggestion level F&Os were related to document enhancement and, according to the peer reviewers, would not affect the screening results. In March 2013, the response to the three suggestion level F&Os was reviewed by the peer review team who had conducted the focused peer review on May 10, 2012. The peer review team concluded that the F&Os were addressed adequately; therefore, they are considered closed.

NRC RAI #5

The NRC has endorsed two methods to accomplish categorization of passive SSC functions to support implementation of 50.69, Revision 0 of American Society of Mechanical Engineers (ASME) Code Case N-660 (N-660) and WCAP-16308-NP-A (WCAP, Reference 7). The LAR refers to these methods and also refers to EPRI-TR-112657 (EPRI), an NRC endorsed methodology to risk-inform inservice inspection (RI-ISI) that is unrelated to 50.69. SNC proposes a new method which uses certain elements (i.e., paragraphs) from each of the three methods. Each endorsed method incorporates all the elements into a single process whereby some non-conservative elements are acceptable based on other conservative elements, and the safety implications of the collective evaluation is judged against the use of the results. Combining disparate elements of previously approved methods into a new method does not provide any basis for acceptability. In Table 1A in the LAR, SNC compares the SNC method to the method in the WCAP. The LAR also includes Table 1B which purports to compare the SNC method to N-660 but which includes cross-references to the WCAP in almost every row. Both Tables 1A and 1B refer extensively to the acceptability of the EPRI method to justify modifications to the WCAP and N-660 methods into SNC's proposed method. The NRC Staff does not accept this approach. The EPRI method is not mentioned in N-660 or Regulatory Guide (RG) 1.201 and it is mentioned only as applicable to RI-ISI in NEI 00-04 (Reference 8). Furthermore, it is mentioned only in one response to RAI 12 (page B-23) in the WCAP.

Please change your passive categorization process to one of the approved methods. Alternatively, please revise Tables 1A and 1B to exclude any reference to N-660, EPRI, or the WCAP methods in the justification and, instead, provide a technical, safety-related argument for each proposed element. Also, as part of the justification for your proposed method, please include a sensitivity study identifying differences in categorization that arise because of the use of your proposed method instead of one of the approved methods.

SNC Response

At the present time, three methods have been approved by NRC to categorize passive components. Each of these methodologies assumes component failure with a probability of 1.0 and then uses a consequence of failure assessment to rank components. These methods are as follows:

1. Revision 0 of American Society of Mechanical Engineers (ASME) Code Case N-660 (N-660), dated July, 2002

2. ANO-2 SER, Safety Evaluation by the Office of Nuclear Reactor Regulation Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals, dated April 22, 2009. [ML090930246]

3. WCAP-16308-NP-A, Revision 0, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program -Categorization Process -Wolf Creek Generating Station," August 2009 (PA-SEE-0027)" [ML092430185 and ML092430186]

SNC has elected to use the categorization method as approved by NRC for ANO-2 and as outlined in the Safety Evaluation Report by the Office of Nuclear Reactor Regulation Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals, dated April 22, 2009. [ML090930246]. As stated in the Vogtle 10 CFR 50.69 LAR, SNC believes that the guidance provided by the selected method is more comprehensive than the guidance provided by the other two methods while still providing sufficiently realistic insights with regard to categorization of passive components. The following information highlights the additional clarity provided by the SNC selected approach:

- The SNC selected process requires that all safety functions supported by a system be completely evaluated as part of that system's categorization, which is consistent with previous risk-informed applications and the intent of N-660, revision 0, while WCAP-16308-NP-A would allow an 'interim' categorization.

- The SNC selected process clearly identifies all relevant configurations that need to be assessed as part of the categorization process (i.e. operating, standby, demand), while the other two methods only provide a general reference (e.g. reference to EPRI TR-112657, Rev B-A in a RAI response or the ASME N-660 Technical Basis Whitepaper)

- The SNC selected process clearly states that operator actions, when credited, need to meet the requirements of NRC approved methodology for ANO-2. For example, where applicable, the likelihood of operator action success <u>and</u> failure are included in the CCDP/CLERP determination, with the highest consequence rank used in the final categorization.

- A spectrum of break sizes needs to be evaluated in the SNC selected process and the one with the highest consequence rank used.

- The SNC process currently limits the application to Class 2 and 3, and non-code class cornponents (i.e., Class 1 is always high-safety-significant (HSS) for passive categorization).

This methodology is contained in the SNC instruction NMP-ES-065-002. SNC procedures related to 50.69 categorization are attached with this letter Enclosure 2). For information, the following table provides a comparison of the guidance provided by the three methodologies and the SNC procedure for each consequence assessment area.

Comparison of Methodologies for Categorizing Passive Components

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
Table 3-1 "Correspondence of Consequence Categories to Numerical Estimate of Conditional Core Damage Probability (CCDP) and Conditional Large Early release Probability (CLERP)"	**Table I-5** "Quantitative Indices for Consequence Categories"	**Table I-5** "Quantitative Indices for Consequence Categories"	Page A-30, **Table I-5** "Quantitative Indices for Consequence Categories"	**Table 5** "Quantitative Indices for Consequence Categories
Table 3-2 "Definition of Consequence Impact Groups and Configuration"	Not provided in the Code Case or the ASME Technical Basis Paper	**Table I-6** "Definition of Consequence Impact Groups and Configuration"	Not provided	**Table 6** "Definition of Consequence Impact Groups and Configuration"
Table 3-3 "General Guidelines for Assigning Consequence Categories to PBFs Resulting in an Initiating Event"	**Table I-1** "Consequence Category for Initiating Event Impact Group"	**Table I-1** "Consequence Category for Initiating Event Impact Group"	Page A-27, **Table I-1** "Consequence Category for Initiating Event Impact Group"	**Table 1** "Consequence Category for Initiating Event Impact Group"
Table 3-4 "A Plant-Specific Example of Assigning Consequence Categories to PBFs Resulting in an Initiating Event"	Not provided in the Code Case or the ASME Technical Basis Paper	Reference provided to TR-112657 Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
Table 3-5 "Guidelines for Assigning Consequence Categories to Pipe Failure resulting in System/Train Loss"	**Table I-2** "Guidelines for Assigning Consequence Categories to Failure Resulting in System or Train Loss"	**Table I-2** "Guidelines for Assigning Consequence Categories to Failure Resulting in System or Train Loss"	Page A-28, **Table I-2** "Guidelines for Assigning Consequence Categories to Failure Resulting in System or Train Loss"	**Table 2** "Guidelines for Assigning Consequence Categories to Failure Resulting in System or Train Loss"
Table 3-6 "Numerical Illustration for Table 3-5, Guidelines for Assigning Consequence Categories to Pipe Failures Resulting in System/Train Loss"	Not provided but as stated in the ASME Technical Basis Paper for N660, r0, "*The EPRI Topical Report (EPRI, 1999) provides further information on the evaluation of the number of backup systems (portions of systems, trains, or portions of trains) available to perform mitigating functions during*	Reference provided to TR-112657 Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	plant events. The quantitative basis for the evaluation (e.g., one full train unavailability being approximately 1E-2) is also discussed in the EPRI Topical Report."			
Table 3-7 "Numerical Illustration for Table 3-5, Guidelines for Assigning Consequence Categories to Pipe Failures Resulting in System/Train Loss – Upper Bound Sensitivity Case"	Not provided but as stated in the ASME Technical Basis Paper for N660, r0, "*The EPRI Topical Report (EPRI, 1999) provides further information on the evaluation of the number of backup systems (portions of systems, trains, or portions of trains) available to perform mitigating*	Reference provided to TR-112657 Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	functions during plant events. The quantitative basis for the evaluation (e.g., one full train unavailability being approximately 1E-2) is also discussed in the EPRI Topical Report."			
Table 3-8 "Numerical Illustration for Table 3-5, Guidelines for Assigning Consequence Categories to Pipe Failures Resulting in System/Train Loss – Lower Bound Sensitivity Case"	Not provided but as stated in the ASME Technical Basis paper for N660, r0, "*The EPRI Topical Report (EPRI, 1999) provides further information on the evaluation of the number of backup systems (portions of systems, trains, or portions of trains) available to*	Reference provided to TR-112657 Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	perform mitigating functions during plant events. The quantitative basis for the evaluation (e.g., one full train unavailability being approximately 1E-2) is also discussed in the EPRI Topical Report."			
Table 3-9 "Frequency of the Challenge: Numerical Values"	Not provided in the Code Case or the ASME Technical Basis Paper	Reference provided to TR-112657, Rev B-A	Not provided	Explicit reference provided to TR-112657 Rev B-A
Table 3-10 "Backup Trains: Unavailability Values"	Not provided but as stated in the ASME Technical Basis paper for N660, r0, "*The EPRI Topical Report (EPRI, 1999) provides further information on the evaluation*	Reference provided to TR-112657, Rev B-A	Not Provided, but as stated in response to RAI #12, page B-21, "The consequence assessment described in Sections I-3.1.1 and I-3.1.2 of Code Case N660 is taken from Code Case N578 "Risk-	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	of the number of backup systems (portions of systems, trains, or portions of trains) available to perform mitigating functions during plant events. The quantitative basis for the evaluation (e.g., one full train unavailability being approximately 1E-2) is also discussed in the EPRI Topical Report."		informed Requirements for Class 1, 2 and 3 Piping, Method B, Section XI, Division 1. Details of the consequence assessment for Code Case N578 are documented in EPRI TR-112657, Rev B-A "Risk-informed Inservice Inspection Evaluation Procedure [ADAMS Accession No. ML013470102]	
Table 3-10 "Backup Trains: Unavailability Values" – Continued (Human Actions as a backup train)	No prescriptive guidance on crediting human actions is provided but as stated in the ASME Technical Basis paper for N660, r0, "*The EPRI Topical*	Requires that when crediting operator action, the likelihood for success and failure will be determined consistent with TR-112657, Rev	Not Provided, but as stated in response to RAI #12, page B-21, "A white paper is prepared for each ASME Code Case that describes the background for the considerations in the	Requires that when crediting operator action, the likelihood for success and failure will be determined consistent with TR-112657, Rev

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	Report (EPRI, 1999) provides further information on the evaluation of the number of backup systems (portions of systems, trains, or portions of trains) available to perform mitigating functions during plant events."	B-A and the scenario that results in the highest consequence ranking shall be used.	Code Case. The white paper for Code Case N660 Revision 0 describes the use of operator actions in the consequence assessment, consistent with the Code Case N578 process and TR-112657, Rev B-A.	B-A and the scenario that results in the highest consequence ranking shall be used.
Figure 3-3 "Heat Removal, Inventory Control, and Long term Heat Removal Safety Functions"	Not provided but as stated in the ASME Technical Basis paper for N660, r0, "*The EPRI Topical Report (EPRI, 1999) provides further information on the evaluation of the number of backup systems (portions of systems, trains, or*	Consistent with TR-112657 Rev B-A (i.e. Figure 3-3), all functions supporting by the system need to be evaluated and the impact of the system's failure of those functions need to assessed and ranked.	No additional information beyond that provided by the Code Case	Consistent with TR-112657 Rev B-A (i.e. Figure 3-3), all functions supporting by the system need to be evaluated and the impact of the system's failure of those functions need to assessed and ranked.

TR-112657 Rev B-A Dated Dec, 1999	**ASME Code Case N-660 Dated July, 2002**	**Approved ANO-2 Relief Request Dated April, 2009**	**WCAP-16308-NP-A Dated Aug, 2009**	**SNC Instruction NMP-ES-065-002**
	portions of trains) available to perform mitigating functions during plant events. The quantitative basis for the evaluation (e.g., one full train unavailability being approximately 1E-2) is also discussed in the EPRI Topical Report."			
Table 3-11 "Example Calibration of System Train Worth for a BWR Pilot Plant	Not provided	Reference provided to TR-112657, Rev B-A	Not provided	Explicit reference provided to TR-112657 Rev B-A
Table 3-12 "Exposure Time: Numerical Values"	Not provided in the Code Case or the ASME Technical Basis Paper	Reference provided to TR-112657, Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A
Equation 3-4 "Numerical Basis for Table 3-5"	No bases provided for the equivalent Code Case table (i.e. Table I-2),	Reference provided to TR-112657, Rev B-A	No additional information beyond that provided by the Code Case	Explicit reference provided to TR-112657 Rev B-A

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	however the ASME Technical Basis paper for N660, r0 states "These CCDP and CLERP ranges are specified in the EPRI Topical Report (EPRI, 1999) and are determined based on the estimates of the total risk associated with the failure."			
Table 3-13 "Guidelines for Assigning Consequence Categories to Combinations of Consequence Impacts"	**Table I-3** "Consequence Categories for Combination Impact Group"	**Table I-3** "Consequence Categories for Combination Impact Group"	Page A-29, **Table I-3** "Consequence Categories for Combination Impact Group"	**Table 3** "Consequence Categories for Combination Impact Group"
Table 3-14 "Example of Guidelines for Assigning Consequence Categories to Pipe Failures Resulting in Increased Potential for an Unisolated LOCA Outside of Containment"	**Table I-4** "Consequence Categories For Failures Resulting in Increased Potential for an	**Table I-4** "Consequence Categories For Failures Resulting in Increased	Page A-29, **Table I-4** "Consequence Categories For Failures Resulting in Increased Potential for an Unisolated	**Table 4** "Consequence Categories For Failures Resulting in Increased

TR-112657 Rev B-A Dated Dec, 1999	ASME Code Case N-660 Dated July, 2002	Approved ANO-2 Relief Request Dated April, 2009	WCAP-16308-NP-A Dated Aug, 2009	SNC Instruction NMP-ES-065-002
	Unisolated LOCA Outside of Containment"	Potential for an Unisolated LOCA Outside of Containment"	LOCA Outside of Containment"	Potential for an Unisolated LOCA Outside of Containment"

NRC RAI #6

How have the fire PRA SSC importance measures been included in the categorization process? Have the categorization sensitivity studies been performed using the fire PRA for the fire scenarios?

SNC Response

When performing trial categorization of systems (Containment Spray system and Chemical and Volume Control system), the importance measures obtained from the fire PRA were included in the categorization process using the fire PRA that was presented to the peer review team in February 2012. The purpose of using this model was to demonstrate the use of fire PRA for categorizing systems. Note that the fire PRA has been refined further since then, but the trial categorization results have not been updated.

The fire PRA importance measures were included in the categorization process in the same fashion as the internal events (including internal flooding) PRA. In addition, an integrated measure of importance was also performed in accordance with NEI 00-04. The integrated measure weights each value in proportion to that hazard model's overall risk metric.

The following information summarizes how fire PRA importance measures have been included in the categorization process and types of sensitivity studies performed using the fire PRA.

Step 1: F-V and RAW importance measures were obtained from the fire PRA model for each basic event used to represent components of a system that was being categorized. This was done for CDF and LERF.

Because a component could have more than one basic event associated with it (e.g., pump fails to start, pump fails to run, pump in maintenance, common cause failure failures), guidance provided in NEI 00-04 Table 5-1 was used to obtain the aggregate value of F-V and maximum value of RAW.

Step 2: Then the following six sensitivities were performed individually using the fire PRA. F-V and RAW importance measures were obtained for each sensitivity run. This was done for CDF and LERF.

- Increase all human error basic events to their 95th percentile value
- Decrease all human error basic events to their 5th percentile value
- Increase all component common cause events to their 95th percentile value
- Decrease all component common cause events to their 5th percentile value
- Set all maintenance unavailability terms to 0.0
- No credit for manual suppression

Step 3: An integrated measure of importance was performed in accordance with Section 5.6 of NEI 00-04.

Step 4: Preliminary High Safety Significant (HSS) components were identified using the following criteria.

PRA Ranking	Criteria
HSS	Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005
HSS	Maximum of component basic event RAW values > 2
HSS	Maximum of applicable common cause basic events RAW values > 20
LSS	Modeled SSCs that do not meet any of the HSS criteria

The FV and RAW values were examined to determine basic events that fall within a 10% buffer zone for each importance measure (that is, 0.0045 for FV importance and 1.80 for RAW importance) for Integrated Decision-making Panel (IDP) consideration as candidate HSS components. In other words, SNC has applied 10% margin to the NEI 00-04 established threshold for F-V and RAW.

Step 5: The initial and overall sensitivities were performed to determine if the increase in CDF and LERF were within permissible limits as outlined in the Regulatory Guide 1.174.

Risk sensitivity study as outlined in the Chapter 8 of NEI 00-04 was performed after the application of qualitative, quantitative, and defense-in-depth considerations. In this risk sensitivity study, the unavailability (if modeled) and unreliability of all Low Safety Significant (LSS) components modeled in the fire PRA of a system that is being categorized is increased by a factor of 3. This has been called as "initial" sensitivity study. The "overall" sensitivity study is performed in the same manner except that PRA modeled LSS components from the systems that were previously categorized were also included. For example, when performing a trial categorization, Chemical and Volume Control System (CVCS) was categorized after the Containment Spray (CS) system. Therefore, the "overall" sensitivity study for the CVCS included PRA-modeled LSS components from the CS system also.

NRC RAI #7

The NRC observed the Integrated Decisionmaking Panel (IDP) deliberation on November 29, 2011. The NRC observations are documented in, "Vogtle Electric Generating Plant, Units 1 and 2, Audit Report For The Process Being Developed To Support A License Amendment Request To Implement Risk Informed Categorization Of Systems, Structures, And Components," (ADAMs Accession number ML12061A245). One observation during the audit was the lack of clarity regarding the response to the qualitative questions described in NEI 00-04 Section 9.2.2. Specifically, several of the qualitative considerations involve a determination as to whether SSCs provides "the sole means" of accomplishing a function. During the audit it was evident that, if loosely applied, no SSC would ever be a "sole means." For example, loss of some radiation monitoring (RM) disabled information relied upon to identify an "adverse containment condition." Depending on the accident scenario other SSCs, such as temperature and/or pressure indicators, could provide an alternative means that would indicate an "adverse condition" and therefore it could be claimed that the RM SSCs do not provide the sole means. Such an evaluation appears not to comply with both the letter and intent of the consideration. Although some scenarios may provide alternative means, some scenarios do not. Furthermore, if the alternative means requires creative interpretation it is not clear that such means would be effective under stressful conditions during an accident unless the alternative is proceduralized and included in the training. At a minimum, any alternative means should be well documented by the IDP. Please provide example documentation of alternative means that have been credited. If any additional guidance beyond that in NEI 00-04 has been developed to provide clarity about "sole means", please provide that guidance.

SNC Response

SNC utilizes the questions in Section 9.2.2 of NEI 00-04 as a means to qualitatively assess the risk of system functions. Refer to SNC procedure NMP-ES-065-003, Section 5.10.1 (Enclosure 2) for details. "Sole means" is used only in the following two questions, as shown in the SNC procedure.

- Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient? This also applies to instrumentation and other equipment needed to allow the required actions to be performed. (EOPA).

- Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities? This also applies to instrumentation and other equipment needed to allow the required actions to be performed. (EOPC)

The response to the questions are based on the entire function, and not on individual components that support the function.

During the November 29, 2011, Integrated Decision-making Panel (IDP) meeting, the IDP members reviewed the categorization of the Containment Spray (CS) system and the Radiation Monitoring (RM) system. One of the IDP comments was that where the responses to the above questions were answered in the negative because the function was not the "sole means", the alternative means had not been specifically identified in the information package provided to the IDP. As a result, the affected responses were revised to provide this additional detail. When performing the subsequent categorization of the Chemical and Volume Control System (CVCS), this lesson learned was applied.

Using a few examples, the information in the tables below illustrates what was provided to the IDP members at the November 29, 2011, meeting and how the response was subsequently revised.

SNC agrees that alternative means should not be credited unless the alternative is proceduralized and included in the training. Although this philosophy was applied during the final categorizations of the three systems, it is not explicitly identified in the SNC procedure. SNC will revise its procedure to ensure that alternative means are not credited as a basis to answer the EOPA or EOPC question as "No", unless the alternative is proceduralized and included in Licensed Operator training. Note that this has been included as Change #4 in RAI #1.

Examples of EOPA and EOPC Questions Answered as "No" (i.e., Not "sole means")

Function ID	Function Description	Legend	Initial Response	Resolution
Radiation Monitoring System				
1.2	PROVIDE CAPABILITY TO MONITOR, INDICATE, AND ALARM PROCESS FLUID RADIOACTIVITY IN THE COMPONENT COOLING WATER (RE□0017A/B).	EOPA	Radiation monitors RE-0017A/B are not used in any EOPs as a sole means of accident mitigation.	RE-0017A/B monitor radiation in the CCW process stream. This is not a parameter monitored in the EOPs as a means of accident mitigation.
		EOPC	Radiation monitors RE-0017A/B are not used in any EOPs as a sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning.	RE-0017A/B monitor radiation in the CCW process stream. This is not a parameter monitored in the EOPs as a means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning
1.3	PROVIDE CAPABILITY TO MONITOR, INDICATE, AND ALARM PROCESS FLUID RADIOACTIVITY IN THE STEAM GENERATOR SAMPLE LIQUID (RE-0019).	EOPA	Radiation monitor RE-0019 is not used in any EOPs as a sole means of accident mitigation. However, it is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. (19000-C, 19010-C, 19020-C, 19100-C, 19121-C & 19233-C)	RE-0019 is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. However, it is not the sole means of accident mitigation. Other means of detecting a SGTR include Chemistry sampling; pressurizer pressure and level lowering; steam generator level rising;

Function ID	Function Description	Legend	Initial Response	Resolution
				onsite and offsite monitoring teams; and MSL or SJAE radiation monitors (prior to isolation).
		EOPC	Radiation monitor RE-0019 is not used in any EOPs as a sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. However, it is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. (19000-C, 19010-C, 19020-C, 19100-C, 19121-C & 19233-C)	RE-0019 is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. However, it is not the sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. Other means of detecting a SGTR include Chemistry sampling; pressurizer pressure and level lowering; steam generator level rising; onsite and offsite monitoring teams; and MSL or SJAE radiation monitors (prior to isolation).
1.5	PROVIDE CAPABILITY TOMONITOR, INDICATE, AND ALARM PROCESS FLUID RADIOACTIVITY IN THE AUXILIARY COMPONENT COOLING WATER (RE□1950).	EOPA	Radiation monitor RE-1950 is not used in any EOPs as a sole means of accident mitigation.	RE-1950 monitors radiation in the ACCW process stream. This is not a parameter monitored in the EOPs as a means of accident mitigation. Leakage into ACCW can also be detected by surge tank level rise and/or thermal barrier high flow and pressure.
		EOPC	Radiation monitor RE-1950 is not used in any EOPs as a sole means of assuring	RE-1950 monitors radiation in the ACCW process stream. This is not a parameter monitored

Function ID	Function Description	Legend	Initial Response	Resolution
			containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning.	in the EOPs as a means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. Leakage into ACCW can also be detected by surge tank level rise and/or thermal barrier high flow and pressure.
1.6	PROVIDE CAPABILITY TO MONITOR, INDICATE, AND ALARM CONTAINMENT ATMOSPHERIC GASEOUS AND PARTICULATE RADIOACTIVITY (RE□2562A/B/C) TO MONITOR CONTAINMENT AIR RADIATION AND TO DETECT REACTOR COOLANT LEAKAGE.	EOPA	Radiation monitors RE-2562A/B/C are not used in any EOPs as a sole means of accident mitigation. However, RE-2562A/C are part of the RCS Leakage Detection System (TS 3.4.15). RE-2562C also provides input to Emergency Classification per NMP-EP-110-GL03.	RE-2562A/C are part of the RCS Leakage Detection System (TS 3.4.15). RE-2562C also provides input to Emergency Classification per NMP-EP-110-GL03. Other RCS leakage detection is provided by Containment sump level, cooler condensate flow rate, containment moisture, containment temperature and containment pressure. Therefore, RE-2562A/B/C are not used as the sole means of accident mitigation.
		EOPC	Radiation monitors RE-2562A/B/C are not used in any EOPs as a sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. However, RE-2562A/C are part of the RCS	RE-2562A/C are part of the RCS Leakage Detection System (TS 3.4.15). RE-2562C also provides input to Emergency Classification per NMP-EP-110-GL03. Other RCS leakage detection is provided by Containment sump level, cooler condensate flow rate,

Function ID	Function Description	Legend	Initial Response	Resolution
			Leakage Detection System (TS 3.4.15). RE-2562C also provides input to Emergency Classification per NMP-EP-110-GL03.	containment moisture, containment temperature, and containment pressure. In addition, leakage via this pathway would flow through the plant vent with its normal and emergency radiation monitors. Therefore RE-2562A/B/C are not used as the sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning.
2.4	PROVIDE CAPABILITY TO MONITOR, INDICATE, AND ALARM RADIOACTIVITY IN THE STEAM GENERATOR BLOWDOWN LIQUID (RE-0021).	EOPA	Radiation monitor RE-0021 is not used in any EOPs as a sole means of accident mitigation. However, it does provide a closure signal to RV-0021 in the event of high radiation in the steam generator blowdown process stream. Also, it is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. (19000-C, 19010-C, 19020-C, 19100-C, 19121-C & 19233-C) RE-0021 also provides input to Emergency Classification per NMP-EP-110-GL03.	RE-0021 is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident and provides a signal to isolate SGBD on high radiation in the effluent stream. However, it is not the sole means of accident mitigation. Other means of detecting a SGTR include Chemistry sampling; pressurizer pressure and level lowering; and steam generator level rising or MSL or SJAE rad monitors prior to isolation. In addition, SGBD is isolated during accident conditions by the AFW actuation signal.
		EOPC	Radiation monitor	RE-0021 is used in

Function ID	Function Description	Legend	Initial Response	Resolution
			RE-0021 is not used in any EOPs as a sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. However, it does provide a closure signal to RV-0021 in the event of high radiation in the steam generator blowdown process stream. Also, it is used in several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident. (19000-C, 19010-C, 19020-C, 19100-C, 19121-C & 19233-C) RE-0021 also provides input to Emergency Classification per NMP-EP-110-GL03.	several EOPs as one of multiple indications of a Steam Generator Tube Rupture (SGTR) accident and provides a signal to isolate SGBD on high radiation in the effluent stream. However, it is not the sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. Other means of detecting a SGTR include Chemistry sampling; pressurizer pressure and level lowering; and steam generator level rising or MSL or SJAE rad monitors prior to isolation. In addition, SGBD is isolated during accident conditions by the AFW actuation signal.
2.7	PROVIDE CAPABILITY TO MONITOR, INDICATE, AND ALARM EFFLUENT RADIOACTIVITY IN THE TURBINE BUILDING DRAIN (RE-0848).	EOPA	Radiation monitor RE-0848 is not used in any EOPs as a sole means of accident mitigation. However, RE-0848 does automatically isolate the turbine building drain path to the retention basin and diverts flow to the turbine building dirty drain tank in the event that the effluent radiation level	RE-0848 monitors and automatically isolates the turbine building drain path in the event that the effluent radiation level exceeds the high limit. RE-0848 also provides an input to Emergency Classification at the Alert and NOUE levels as one threshold input along with sampling and release rate

Function ID	Function Description	Legend	Initial Response	Resolution
			exceeds the high limit. RE-0848 also provides an input to Emergency Classification per NMP-EP-110-GL03.	calculations (as confirmed by the ODCM). TB drain activity is confirmed by sampling, and the flow path can be manually isolated. In addition, the discharge path is to the waste water retention basin which provides significant holdup and opportunity to mitigate the release prior to flowing offsite. Therefore, RE-0848 is not the sole means of accident mitigation.
		EOPC	Radiation monitor RE-0848 is not used in any EOPS as a sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning. However, RE-0848 does automatically isolate the turbine building drain path to the retention basin and diverts flow to the turbine building dirty drain tank in the event that the effluent radiation level exceeds the high limit. RE-0848 also provides an input to Emergency Classification per NMP-EP-110-GL03.	RE-0848 monitors and automatically isolates the turbine building drain path in the event that the effluent radiation level exceeds the high limit. This entire system is outside containment and has no impact on containment integrity. RE-0848 also provides an input to Emergency Classification at the Alert and NOUE levels as one threshold input along with sampling and release rate calculations (as confirmed by the ODCM). TB drain activity is confirmed by sampling, and the flow path can be manually isolated. Therefore, RE-0848

Function ID	Function Description	Legend	Initial Response	Resolution
				is not the sole means of assuring containment integrity, monitoring of Post-Accident conditions, or offsite emergency planning.
Containment Spray System				
4.1	PROVIDE CONTAINMENT PRESSURE SIGNALS AS REQUIRED TO SUPPORT SOLID STATE PROTECTION SYSTEM (SSPS).	EOPA	Containment pressure signals through SSPS provide input to actuations that are verified in the EOP's, i.e. SI and CS. In the case of SI, other parameters provide input to SSPS; in addition, these automatic signals are not the "sole means" for successful completion, since the EOP's also rely on manual operator actions based on instrumentation and alarms to back up automatic actuations.	Containment pressure signals through SSPS provide input to actuations that are verified in the EOP's, i.e. SI and CS. In the case of SI, other parameters provide input to SSPS including low pressurizer pressure, low steam pressure and manual actuation; in addition, these automatic signals are not the "sole means" for successful completion, since the EOP's also rely on manual operator actions based on instrumentation and alarms to back up automatic actuations.
		EOPC	Containment pressure signals through SSPS provide input to actuations that are verified in the EOP's, i.e. SI and CS. In the case of SI, other parameters provide input to SSPS; in addition, these	Containment pressure signals through SSPS provide input to actuations that are verified in the EOP's, i.e. SI and CS. In the case of SI, other parameters provide input to SSPS including low pressurizer pressure,

Function ID	Function Description	Legend	Initial Response	Resolution
			automatic signals are not the “sole means” of achieving actions, since the EOP’s also rely on manual operator actions based on instrumentation and alarms to back up automatic actuations	low steam pressure and manual actuation; in addition, these automatic signals are not the “sole means” for successful completion, since the EOP’s also rely on manual operator actions based on instrumentation and alarms to back up automatic actuations.
Chemical and Volume Control System (CVCS)				
1	Maintains primary coolant inventory during normal operations, startup, and shutdown (includes operation in support of accident response when restoring CVCS inventory control).	EOPA	N/A (lesson learned from CS and RM was applied)	Restoration of PZR level following a trip or a LOCA event is called out in EOPs, but is not the sole means of maintaining inventory control. Other means such as depressurization and injection via Safety Injection is available.
		EOPC	N/A (lesson learned from CS and RM was applied)	CVCS does not provide for or contribute to long term containment integrity, post accident monitoring, or offsite planning activities. Alternate means of inventory control include cooldown, depressurization and makeup via the SI system.
3	Controls primary coolant pH during normal operations, startup, and shutdown.	EOPA	N/A (lesson learned from CS and RM was applied)	Control of primary coolant chemistry parameters is called out in the AOPs, but is not required to mitigate an accident

Function ID	Function Description	Legend	Initial Response	Resolution
				or transient.
		EOPC	N/A (lesson learned from CS and RM was applied)	Control of primary coolant chemistry parameters is called out in the AOPs, but is not required to ensure containment integrity, post accident monitoring, or offsite planning activities.
7	Provides seal water injection for RCP seal cooling and integrity.	EOPA	N/A (lesson learned from CS and RM was applied)	This function is called out in the EOPs but is not the sole means of mitigating transient or accident initiators. RCP thermal barrier cooling is the alternate means of success if seal injection fails.
		EOPC	N/A (lesson learned from CS and RM was applied)	Loss of seal injection has no impact on containment integrity, post accident monitoring, or offsite planning activities.
10	Provides auxiliary spray for pressure reduction when normal sprays are not available.	EOPA	N/A (lesson learned from CS and RM was applied)	Auxiliary sprays are called out in the EOPs (such as depressurizing the RCS to prevent over pressurizing a steam generator with a tube rupture), but is not the sole means for reducing RCS pressure. PORV operation can provide this capability.
		EOPC	N/A (lesson learned from CS and RM was applied)	Auxiliary sprays are called out in the EOPs (such as depressurizing the RCS to prevent over pressurizing a steam generator with a tube rupture), but is not

Function ID	Function Description	Legend	Initial Response	Resolution
				the sole means for reducing RCS pressure. PORV operation can provide this capability.

Examples of EOPA and EOPC Questions Answered as "Yes" (i.e., "sole means" only)

Function ID	Function Description	Legend	Initial Response	Resolution
Containment Spray System				
4.2	PROVIDE CONTAINMENT PRESSURE INDICATION AS REQUIRED TO SUPPORT THE POST ACCIDENT MONITORING SYSTEM.	EOPA	Post accident monitoring instruments are used in the EOP's to trigger operator actions, but these actions are a backup to signals through SSPS which input to automatic actuations. Therefore this function is not the "sole means" for successful performance.	Post accident monitoring instruments for containment pressure are the sole means of determining an adverse containment condition due to high containment pressure.This parameter is used throughout the EOPs to trigger operator actions required to mitigate an accident or transient.
		EOPC	Containment pressure instruments are used in the EOP's as the sole means for Post accident monitoring of this parameter. Containment pressure monitoring also affects the emergency plan, discriminating between a Site Area and General Emergency.	Containment pressure instruments are used in the EOP's as the sole means for Post accident monitoring of this parameter. Containment pressure monitoring also affects the emergency plan, discriminating between a Site Area and General Emergency.
Chemical and Volume Control System (CVCS)				
8	Provides Head Vent for removing steam and non-condensable gases during accident conditions (EOP	EOPA	N/A	EOP 19263-C calls for use of head vent for removal of steam and noncondensable gases during accident conditions. That head vent flow path is the

Function ID	Function Description	Legend	Initial Response	Resolution
	19263-C).			sole means of removing those gases from the RPV head once they form during an accident condition.
		EOPC	N/A	Failure of the head vent function would not impact containment integrity, post accident monitoring, or offsite planning activities.
9	Provides emergency boration (EB) for ATWS events or boron dilution events	EOPA	N/A	EOP 19001-C calls for using emergency boration in the event of an ATWS event. Also referred to EOP 19211-C or 19212-C from EOP 19200-C (critical safety function status trees). Failure of EB is the sole means of shutting down the reactor following an ATWS event where the control rods cannot be inserted.
		EOPC	N/A	Failure of emergency boration would not impact the ability to maintain containment integrity, post accident monitoring, or offsite planning activities.

NRC RAI #8

Please summarize the risk sensitivity study described in Chapter 8 of NEI 00-04. Please include the unreliability factor selected and the change in both the internal events and fire risk metrics upon use of the factor.

SNC Response

This RAI is answered using the results from the trial categorization.

A trial categorization of three systems - Containment Spray (CS) system, Radiation Monitoring (RM) system, and Chemical and Volume Control System (CVCS) – was performed prior to submitting the Vogtle 10 CFR 50.69 LAR. Because the RM system is not logically modeled in the PRA, the sensitivity study described in Chapter 8 of NEI 00-04 would not be applicable. Hence, the following information summarizes the risk sensitivity study performed for two systems – CS and CVCS. As outlined on Page E1-20 of the LAR, a factor of 3 has been used when performing the sensitivity study described in Chapter 8 of NEI 00-04.

CS System:

Per Chapter 8 of NEI 00-04, the initial and overall sensitivities are performed to determine if the increase in CDF and LERF are within permissible limits as outlined in the Regulatory Guide 1.174.

Initial Sensitivity Study:

The initial sensitivity study applies to a system that is being categorized (in this case CS system). In this initial sensitivity study, the failure probability of all LSS components belonging to CS system and modeled in the Internal Events (including Internal Flooding) and Fire PRAs are increased by a factor of 3. The following components were categorized as LSS in the CS system and are modeled in the PRAs. All other PRA modeled components for the CS system were categorized as HSS because of quantitative risk assessment, qualitative risk assessment, or application of defense-in-depth considerations.

Component ID	Basic Event ID in PRA Model	3x Probability	Probability in Baseline PRA	Description of Basic Event
11206P6001	1CSPM001-----A	3.27E-03	1.09E-03	MDP 001 FAILS TO START DUE TO RANDOM FAULTS
11206P6001	1CSPM001-----X	1.53E-03	5.11E-04	MDP 001 FAILS TO RUN DUE TO RANDOM FAILURE
11206P6001	1CSPM001-----M	5.34E-03	1.78E-03	MDP 001 MAINT. UNAVAILABILITY - INLUDES MOTOR OPERATED VALVES
11206P6001	1CSPM001002--ACC	2.03E-04	6.78E-05	MDP 001 AND 002 FAIL TO START DUE TO COMMON CAUSE
11206P6001	1CSPM001002--XCC	5.34E-05	1.78E-05	MDP 001 AND 002 FAIL TO RUN DUE TO COMMON CAUSE
11206P6002	1CSPM002-----A	3.27E-03	1.09E-03	MDP 002 FAILS TO START DUE TO RANDOM FAULTS
11206P6002	1CSPM002-----X	1.53E-03	5.11E-04	MDP 002 FAILS TO RUN DUE TO RANDOM FAILURE
11206P6002	1CSPM002-----M	5.34E-03	1.78E-03	MDP 002 MAINT. UNAVAILABILITY - INCLUDES MOTOR OPERATED VALVES
11206P6002	1CSPM001002--ACC	2.03E-04	6.78E-05	MDP 001 AND 002 FAIL TO START DUE TO COMMON CAUSE
11206P6002	1CSPM001002--XCC	5.34E-05	1.78E-05	MDP 001 AND 002 FAIL TO RUN DUE TO COMMON CAUSE
11206U6001	1CSCV1206-001K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT
11206U6008	1CSCV1206-008K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT
11206U6037	1CSCV1208-037K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT
11206U6038	1CSCV1208-038K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT

After increasing the probability (unreliability and unavailability, as appropriate) of the above mentioned components, the Internal Events (including Internal Flooding) and Fire

PRA models were quantified. The following table summarizes CDF and LERF. As shown in the following table, the delta CDF and delta LERF are less that 1E-06.

	IE (including IF)		Fire	
	CDF	LERF	CDF	LERF
Baseline Model	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Increase UR and UA 3 times	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Delta	0.00E+00	1.90E-11	0.00E+00	0.00E+00

Cumulative Sensitivity Study:

In the cumulative sensitivity study, the probability (unreliability and unavailability, as appropriate) of all LSS components for systems that have been categorized and modeled in PRAs is increased by a factor of 3. Because the CS system is the first system that was categorized, the cumulative sensitivity study was not applicable as no other systems had been categorized previously.

CVCS System:

CVCS was categorized after categorizing CS system.

Initial Sensitivity Study:

The initial sensitivity study applies to a system that is being categorized (in this case CVCS system). In this initial sensitivity study, the failure probability of all LSS components belonging to CVCS system and modeled in the Internal Events (including Internal Flooding) and Fire PRAs are increased by a factor of 3. The following is the only component that was categorized as LSS and is modeled in the PRAs. All other PRA modeled components were categorized as HSS because of quantitative risk assessment, qualitative risk assessment, or application of defense-in-depth considerations.

Component ID	Basic Event ID in PRA Model	3x Probability	Probability in Baseline PRA	Description of Basic Event
1HV8924	1HPMVHV8924--P	7.59E-04	2.53E-04	MOV HV8924 IN CCP and SIP SUCTION X-CONNECTION PLUGS

After increasing the failure probability of the above component, the Internal Events (including Internal Flooding) and Fire PRA models were quantified. The following table summarizes CDF and LERF. As shown in the following table, the delta CDF and delta LERF are less that 1E-06.

CVCS Only

	IE (including IF)		Fire	
	CDF	LERF	CDF	LERF
Baseline Model	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Increase UR and UA 3 times	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Delta	0.00E+00	1.90E-11	0.00E+00	0.00E+00

<u>Cumulative Sensitivity Study</u>:

In the cumulative sensitivity study, the probability (unreliability and unavailability, as appropriate) of all LSS components for systems that have been categorized and modeled in PRAs is increased by a factor of 3. Therefore, probability (unreliability and unavailability, as appropriate) of all LSS PRA modeled components from the CS system and CVCS is increased by a factor of 3. The following table summarizes unreliability and unavailability events for the CS system and CVCS for which probability (unreliability and unavailability, as appropriate) of all LSS components modeled in the PRAs was increased by a factor of 3.

Component ID	Basic Event ID in PRA Model	3x Probability	Probability in Baseline PRA	Description of Basic Event (System)
11206P6001	1CSPM001-----A	3.27E-03	1.09E-03	MDP 001 FAILS TO START DUE TO RANDOM FAULTS (CS)
11206P6001	1CSPM001-----X	1.53E-03	5.11E-04	MDP 001 FAILS TO RUN DUE TO RANDOM FAILURE (CS)
11206P6001	1CSPM001-----M	5.34E-03	1.78E-03	MDP 001 MAINT. UNAVAILABILITY - INLUDES MOTOR OPERATED VALVES (CS)
11206P6001	1CSPM001002--ACC	2.03E-04	6.78E-05	MDP 001 AND 002 FAIL TO START DUE TO COMMON CAUSE (CS)
11206P6001	1CSPM001002--XCC	5.34E-05	1.78E-05	MDP 001 AND 002 FAIL TO RUN DUE TO COMMON CAUSE (CS)
11206P6002	1CSPM002-----A	3.27E-03	1.09E-03	MDP 002 FAILS TO START DUE TO RANDOM FAULTS (CS)
11206P6002	1CSPM002-----X	1.53E-03	5.11E-04	MDP 002 FAILS TO RUN DUE TO RANDOM FAILURE (CS)
11206P6002	1CSPM002-----M	5.34E-03	1.78E-03	MDP 002 MAINT. UNAVAILABILITY - INCLUDES MOTOR OPERATED VALVES (CS)
11206P6002	1CSPM001002--ACC	2.03E-04	6.78E-05	MDP 001 AND 002 FAIL TO START DUE TO COMMON CAUSE (CS)
11206P6002	1CSPM001002--XCC	5.34E-05	1.78E-05	MDP 001 AND 002 FAIL TO RUN DUE TO COMMON CAUSE (CS)

11206U6001	1CSCV1206-001K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT (CS)
11206U6008	1CSCV1206-008K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT (CS)
11206U6037	1CSCV1208-037K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT (CS)
11206U6038	1CSCV1208-038K	1.52E-03	5.06E-04	CTMT SPRAY CV FAILS TO SEAT (CS)
1HV8924	1HPMVHV8924--P	7.59E-04	2.53E-04	MOV HV8924 IN CCP and SIP SUCTION X-CONNECTION PLUGS (CVCS)

As shown in the following table, the delta CDF and delta LERF are less than 1E-06.

CS and CVCS

	IE (including IF)		Fire	
	CDF	**LERF**	**CDF**	**LERF**
Baseline Model	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Increase UR and UA 3 times	2.25E-05	7.37E-08	5.30E-05	1.45E-06
Delta	0.00E+00	1.90E-11	0.00E+00	0.00E+00

NRC RAI #9

The LAR reported that the Peer Review identified 36 fire PRA supporting requirements that were not met or Capability Category (CC) I excluding 25 deemed to be not applicable. The LAR concludes that all SRs aside from 2 documentation Supporting Requirements (SRs) are currently being met at CC II or better, and that 2 additional SRs are satisfactorily met at CC I. Please:

a. Summarize the review process and the qualifications of the personnel that have reviewed your resolutions to determine the post-resolution category of each SR.
b. Clarify whether the peer review team or another party deemed the 25 SRs inapplicable.
c. Summarize the 25 SRs deemed not applicable and provide the criteria used to make that determination.

SNC Response

a. The Peer Review of the Vogtle Electric Generating Plant (VEGP) Fire Probabilistic Risk Assessment (FPRA) was performed against the requirements of Section 4 of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009, Addenda A to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME and the American Nuclear Society, December 2008) and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200, Revision 2. This peer review was performed using the process defined in Nuclear Energy Institute (NEI) 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Nuclear Energy Institute, November 2008).

The scope of the peer review was against all technical elements in Section 4 of the ASME/ANS PRA Standard with the exception of the Qualitative and Quantitative Screening technical elements (QLS and QNS). It was conducted the week of February 13 through February 17, 2012.

The peer review team consisted of eight (8) reviewers having a full range of experience required to perform the peer review. In addition, two NRC staff members were present as silent observers during the peer review including consensus sessions. Each reviewer was assigned a lead role for a review element. The documents were supplied in advance to the peer review team members. During the week of February 13 through February 17, 2012, the peer review team was onsite, performed the review, and provided preliminary results on February 17, 2012. A final report was issued on August 28, 2012. The following table summarizes results of the peer review.

Summary of Overall Results of the VEGP Fire PRA Peer Review										
Fire PRA Element	Number of Supporting Requirements Meeting Each Capability Category									
	Not Met	Met	CC-I	CC-I/II	CC-II	CC-II/III	CC-III	Not Applicable	Not Reviewed	Total
PP	5	7	0	0	0	0	0	0	0	12
ES	1	6	0	0	1	0	4	2	0	14
CS	0	10	0	0	0	1	2	3	0	16
QLS*	0	0	0	0	0	0	0	7	0	7
PRM	2	15	0	0	0	0	0	3	0	20
FSS	20	16	4	4	0	5	0	1	0	50
IGN	0	10	0	0	2	0	1	2	0	15
QNS*	0	0	0	0	0	0	0	6	0	6
CF	0	2	0	0	0	1	0	0	0	3
HRA	0	5	1	1	4	1	0	0	0	12
SF	0	6	0	0	0	0	0	0	0	6
FQ	2	7	0	0	0	0	0	1	0	10
UNC	0	2	0	0	0	0	0	0	0	2
MU	1	9	0	0	0	0	0	0	0	10
TOTALS	31	95	5	5	7	8	7	25	0	183

* SNOC stated that they did not do qualitative or quantitative screening so the requirements of QLS and QNS are not applicable. However there was evidence to the contrary. Such instances were cited under the specific technical element.

QLS and QNS were subsequently evaluated under a focused scope peer review. See response for RAI #4.

Qualifications of the personnel that reviewed the F&O resolutions are similar with those qualifications specified for the peer reviewers in ASME/ANS RA-Sa-2009 Standard Section 1-6.2. That is, the personnel reviewing the F&O resolutions are knowledgeable of the standard requirements and are experienced in performing the applicable PRA activities.

b. The peer review team assessed 25 SRs as not applicable.

c. Table 1 of this RAI response summarizes the 25 SRs deemed not applicable by the peer review team. The table provides the SR designator, the CC II requirement, and the peer review team assessment from Appendix B of the peer review report.

As discussed in the LAR, a focused scope peer review was performed for the QLS and QNS elements and resulted in the seven SRs for QLS being assessed as MET and the six SRs for QNS as NA (not applicable). These assessments are not reflected in Table 1, mentioned below.

Table 1
Summary of the SRs Assessed as Not Applicable

Entry	SR	CC II Requirement	Peer Review Team Assessment
1	ES-B3	INCLUDE additional equipment if that equipment is associated with new initiating events or different accident sequences that go beyond that treated within the scope of either or both the Fire Safe Shutdown/Appendix R work or the Internal Events PRA with a potential for being a significant contributor to the CDF/LERF in the Fire PRA.	No new accident sequences or new initiators are included in the model.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
2	ES-B5	EXCLUDE, if desired, equipment or failure modes from identification and inclusion in the Fire PRA based on the following: *(a)* a fire-induced spurious operation of a component may be excluded from a system model if the conditional probability of occurrence given fire-induced damage to the component and/or associated cables is at least two orders of magnitude lower than the non-fire-induced random failure probability of the other components in the same system train that results in the same effect on system operation. The justification for exclusion must include the consideration of the scope of potential fire-induced failures to the system/train under consideration that may reasonably occur. *(b)* one or more fire-induced spurious operations of components may be excluded from the systems model if the contribution of their conditional probability of occurrence given fire-induced damage to them and/or their associated cables is <1% of the total failure rate or probability for that component or group of components, when their effects on system operation are the same. The justification for exclusion must include the consideration of the scope of potential fire-induced failures to the system/train under consideration that may reasonably occur.	The screening criteria specified in ES-B5 were not used in the PRA analysis.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
3	CS-A7	For ungrounded power distribution systems for three-phase-powered equipment that could spuriously operate due to proper polarity intercable hot shorts, INCLUDE these cable and circuit failure modes in the Fire PRA plant response model to the extent that a spurious operation of a single piece of equipment might lead to an interfacing system LOCA or containment bypass that results in core damage and large early release.	Analysis methodology described in PRA-BC-V-12-005, Attachment 1, Section 6.3.3.2. Conversation with circuit analysis staff indicated no circuits of this type were analyzed, typically delta (vs. wye) three-phase systems.
4	CS-A8	IDENTIFY instances where thermoplastic insulated power supply circuits are applied *and* INCLUDE the treatment of cable failures involving three-phase-powered equipment that could spuriously operate and lead to an interfacing system LOCA or containment bypass that results in core damage and large early release due to a proper polarity three-phase hot short.	Analysis methodology described in PRA-BC-V-12-005, Attachment 1, Section 6.3.3.2. Conversation with circuit analysis staff indicated no circuits of this type were analyzed; the majority of cables (~95%) were found to have thermoset insulation by analysis of cables listed in PDMS.
5	CS-C3	If the provision of SR CS-A11 is used, DOCUMENT the assumed cable routing and the basis for concluding that the routing is reasonable in a manner that facilitates Fire PRA applications, upgrades, and peer review.	Cable routing was not assumed.
6	QLS-A1	RETAIN for quantitative analysis those physical analysis units that contain equipment or cables required to ensure as-designed circuit operation, or whose failure could cause spurious operation, of any equipment, system, function, or operator action credited in the Fire PRA plant response model.	VEGP did not use Qualitative Screening.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
7	QLS-A2	RETAIN for quantitative analysis those physical analysis units where a fire might require a manual or automatic plant trip or a controlled manual shutdown based on plant Technical Specifications *and* If a time limit is established for a required Technical Specifications required shutdown, ESTABLISH a basis for the applied time window.	VEGP did not use Qualitative Screening.
8	QLS-A3	APPLY the screening criteria to each physical analysis unit defined in the plant partitioning analysis.	VEGP did not use Qualitative Screening.
9	QLS-A4	If additional qualitative screening criteria are applied, DEFINE the applied criteria and PROVIDE A BASIS that shows the applied criteria provide reasonable assurance that physical analysis units that are screened out are negligible contributors to fire risk in a manner consistent, at a minimum, with SRs QLS-A1, QLS-A2, and QLS-A3.	VEGP did not apply additional screening requirements.
10	QLS-B1	DOCUMENT the qualitative screening criteria applied.	VEGP did not use Qualitative Screening.
11	QLS-B2	DOCUMENT the disposition of each physical analysis unit defined by the plant partitioning analysis as either "screened out" or "retained for quantitative analysis" and in a manner that facilitates Fire PRA applications, upgrades, and peer review.	VEGP did not use Qualitative Screening.
12	QLS-B3	DOCUMENT the exclusion basis for each physical analysis unit defined in the plant partitioning analysis that has been screened out in a manner that facilitates Fire PRA applications, upgrades, and peer review.	VEGP did not use Qualitative Screening.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
13	PRM-B6	MODEL accident sequences for any new initiating events identified per PRM-B3 and any accident sequences identified per PRM-B5 reflective of the possible plant responses to the fire-induced initiating events in accordance with HLR-AS-A and HLR-AS-B and their SRs in Part 2 with the following clarifications, *and* DEVELOP a defined basis to support the claim of nonapplicability of any of the following requirements in Part 2: *(a)* All the SRs under HLR-AS-A and HLR-AS-B in Part 2 are to be addressed in the context of fire scenarios including effects on equipment, associated cabling, operator actions, and accident progression and timing. *(b)* When applying AS-A5 in Part 2 to Fire PRA, INCLUDE consideration of fire response procedures as well as emergency operating procedures and abnormal procedures.	No new initiating events specific to the Fire PRA were identified so this SR is N/A.
14	PRM-B7	IDENTIFY any cases where new or modified success criteria will be needed to support the Fire PRA consistently with the HLR-SC-A and HLR-SC-B of Part 2 and their supporting requirements.	Since no new fire-related initiating events or accident sequences were identified, no new or modified success criteria was required, and this SR is considered N/A.
15	PRM-B8	For any cases identified per PRM-B7, CONSTRUCT the Fire PRA plant response model using success criteria that are defined in accordance with HLR-SC-A and HLR-SC-B and their SRs in Part 2 *and* DEVELOP a defined basis to support the claim of nonapplicability of any of these requirements in Part 2.	No new success criteria scenarios were identified, so new logic was not required to model it, and this SR is N/A.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
16	FSS-C8	If raceway fire wraps are credited, *(a)* ESTABLISH a technical basis for their fire-resistance rating, *and* *(b)* CONFIRM that the fire wrap will not be subject to either mechanical damage or direct flame impingement from a high-hazard ignition source unless the wrap has been subject to qualification or other proof of performance testing under these conditions.	Wraps are not credited in the analysis.
17	IGN-A2	Except as allowed by SR IGN-A3, USE applicable data from nonnuclear power industry sources only when there is no similar experience in the nuclear power industry *and* JUSTIFY all nonnuclear power industry sources used for establishing fire ignition frequencies by demonstrating the applicability of information provided in those sources to the specific ignition source being studied *and* In justifying the use of nonnuclear power industry data, INCLUDE verification that applicable nuclear industry data do not exist, a description of the data being applied including its source, discussion of the data analysis approach and methods used to estimate per reactor-year fire frequencies, and verification of the applicability of the applied data to nuclear power plant conditions and the fire scenario(s) being analyzed.	Only nuclear power industry data was used. See Section 3.1 of Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6).

Table 1
Summary of the SRs Assessed as Not Applicable

Entry	SR	CC II Requirement	Peer Review Team Assessment
18	IGN-A3	In cases where nuclear power industry and nonnuclear industry data are not available, USE engineering judgment.	Only nuclear power industry data was used. See Section 3.1 of Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6).
19	QNS-A1	DEFINE quantitative screening criteria that ensure that the cumulative impact of screened physical analysis units on CDF and LERF is small.	Quantitative Screening Not Used.
20	QNS-B1	APPLY the quantitative screening criteria to each physical analysis unit defined by the plant partitioning analysis not previously screened out qualitatively.	Quantitative Screening Not Used.
21	QNS-B2	RETAIN for risk quantification or scenario development each physical analysis unit that does not meet the defined quantitative screening criteria.	Quantitative Screening Not Used.
22	QNS-C1	VERIFY that *(a)* the quantitative screening process does not screen the highest risk fire areas And *(b)* the sum of the CDF contributions for all screened fire compartments is < 10% of the estimated total CDF for internal events And *(c)* the sum of the LERF contributions for all screened fire compartments is < 10% of the estimated total LERF for internal events	Quantitative Screening Not Used.

Table 1 Summary of the SRs Assessed as Not Applicable			
Entry	SR	CC II Requirement	Peer Review Team Assessment
23	QNS-D1	DOCUMENT the disposition per QNS-B of each physical analysis unit defined by the plant partitioning analysis as either screened out or retained for quantitative analysis, and the cumulative impact of the quantitative screening per QNS-C in a manner that facilitates Fire PRA applications, upgrades, and peer review.	Quantitative Screening Not Used.
24	QNS-D2	DOCUMENT the CDF and LERF values used for quantitative screening and the cumulative impact of quantitative screening, for each physical analysis unit defined in the plant partitioning analysis that has been screened out in a manner that facilitates Fire PRA applications, upgrades, and peer review.	Quantitative Screening Not Used.
25	FQ-F2	Document any defined bases to support the claim of nonapplicability of any of the referenced requirements in Part 2 beyond that already covered by the clarifications in this Part.	(None provided. The PRA did not claim non applicability of any of the referenced requirements in Part 2.)

NRC RAI #10

Regarding errors identified in the analyses, the peer review team identified a number of individual errors in the fire PRA evaluation (PRM-A1-01, IGN-B1-01, FQ-C1-02, FSS-B2-02, and FSS-C4-02). The SNC resolution for the indicated F&Os state that the errors were confirmed but isolated to those identified by the peer review. Did SNC's review of the analyses for similar errors include all such potential errors such that there is confidence that the peer review did indeed identify the only errors in the PRA? Or was SNC's review limited to a sample that would provide less confidence that all similar errors had been identified and fixed? Please clarify and justify the process used to review the analyses cited by each F&O listed above in which an error was identified.

SNC Response

F&Os PRM-A1-01, IGN-B1-01, FQ-C1-02, FSS-B2-02, and FSS-C4-02 were discussed and reviewed in detail with the peer review team during the peer review to determine the depth of each identified error in the PRA. In each of these cases, it was demonstrated to be isolated instances resulting in these specific F&Os. The resolutions to F&Os PRM-A1-01, IGN-B1-01, FQ-C1-02, FSS-B2-02, and FSS-C4-02 included a complete review of the PRA based on the insights obtained during the peer review. The purpose of the reviews was to established confidence in the PRA. The steps taken during these reviews included understanding the technical basis of the F&O, identifying and resolving the specific error presented in the F&O, and reviewing the PRA model subject to the identified error for correctness.

NRC RAI #11

Regarding documentation requiring modification, the peer review team identified numerous instances where PRA documentation was confusing, missing, or incomplete (ES-D1-01, CS-C2-01, CS-C2-02, PRM-B13, PRM-C1-01, FSS-A3-01, FQ-F1-01, FQ-F1-02, IGN-A7-01, HRA-B3-01, MU-C1-01, UNC-A2-02, and MU-C1-01). SNC's response was generally that the documentation has been or will be improved. Since 50.69 categorization is performed over many years, proper documentation is needed to provide confidence that PRA updates and actual categorization evaluations appropriately reflect the operation and design of the facility. Please summarize SNC's process to ensure that the documentation of the PRA is now of sufficient clarity and quality to support the long-term, continuous use of the PRA.

SNC Response

The ASME/ANS RA-Sa-2009 Standard includes documentation requirements for each technical element. It requires that the PRA is documented in a manner that facilitates PRA applications, upgrades, and peer reviews. The PRA documentation process includes preparation, review, and approval to ensure that the PRA analysis is sufficiently documented to support continuous use. The peer review process provides the benefit of externally qualified personnel to provide insight into the sufficiency of documentation. The peer review resulted in recommendations to improve the documentation in the referenced F&Os in this RAI. The recommendations in the F&Os from the peer review were included in the updated documentation. Additionally, during the course of the PRA update additional documentation items were improved as the documentation process of preparation, review, and approval was completed. The SNC PRA configuration and control process ensures that the PRA and documentation is sufficient to support the long term continuous use of the PRA.

NRC RAI #12

ASME RA-Sa-2009 SR IE-A5 requires a structured approach (such as a system-by-system review of initiating event potential, a Failure Modes and Effects Analysis or a fault tree) to assess and document the possibility of an initiating event arising from individual system or train failures. Support systems are within the scope of this evaluation.

Initiating events resulting from multiple failures are to be included if the equipment failures result from a common cause.

F&O IE-A4-01 appears to refer to the requirements in IE-A5, not IE-A4. The F&O states that simply crediting an evaluation performed for the Individual Plant Examination (IPE) is not sufficient to demonstrate a "structured approach." Instead, the IPE evaluation should be reviewed and evaluated to determine whether it complies with ASME RA-Sc-2009. The resolution to this F&O states that an evaluation was performed during the Vogtle IPE using a "block diagram" but does not describe the methodology or whether it includes support systems and accounts for common cause failures as called for by ASME RA-Sc-2009. Please describe the structured approach that was used and an explanation of how it was reviewed and found to have met the aforementioned requirements.

SNC Response

A systematic search was performed in identifying initiating events that need to be included in the current VEGP internal event PRA model.

Originally, during VEGP Individual Plant Examination (IPE), internal initiating events were identified through a systematic and comprehensive review of other PRAs such as WASH-1400; NUREG/CR-3862; VEGP FSAR Chapter 15 Category III and IV events; and VEGP system information. In addition, a systematic review of the effects of failures in all VEGP supporting systems was performed in order to identify supporting systems failures which needed to be considered in the Vogtle IPE as special initiating events.

In updating to the current VEGP internal event PRA model (that was peer reviewed in May 2009), a systematic identification of initiating events was performed again. The following Figure 2.1.1, " Initiating Event Update Tasks for the current VEGP PRA Model" ("block diagram"), summarizes a structured approach used to identify internal initiating events for the current VEGP internal events PRA model.

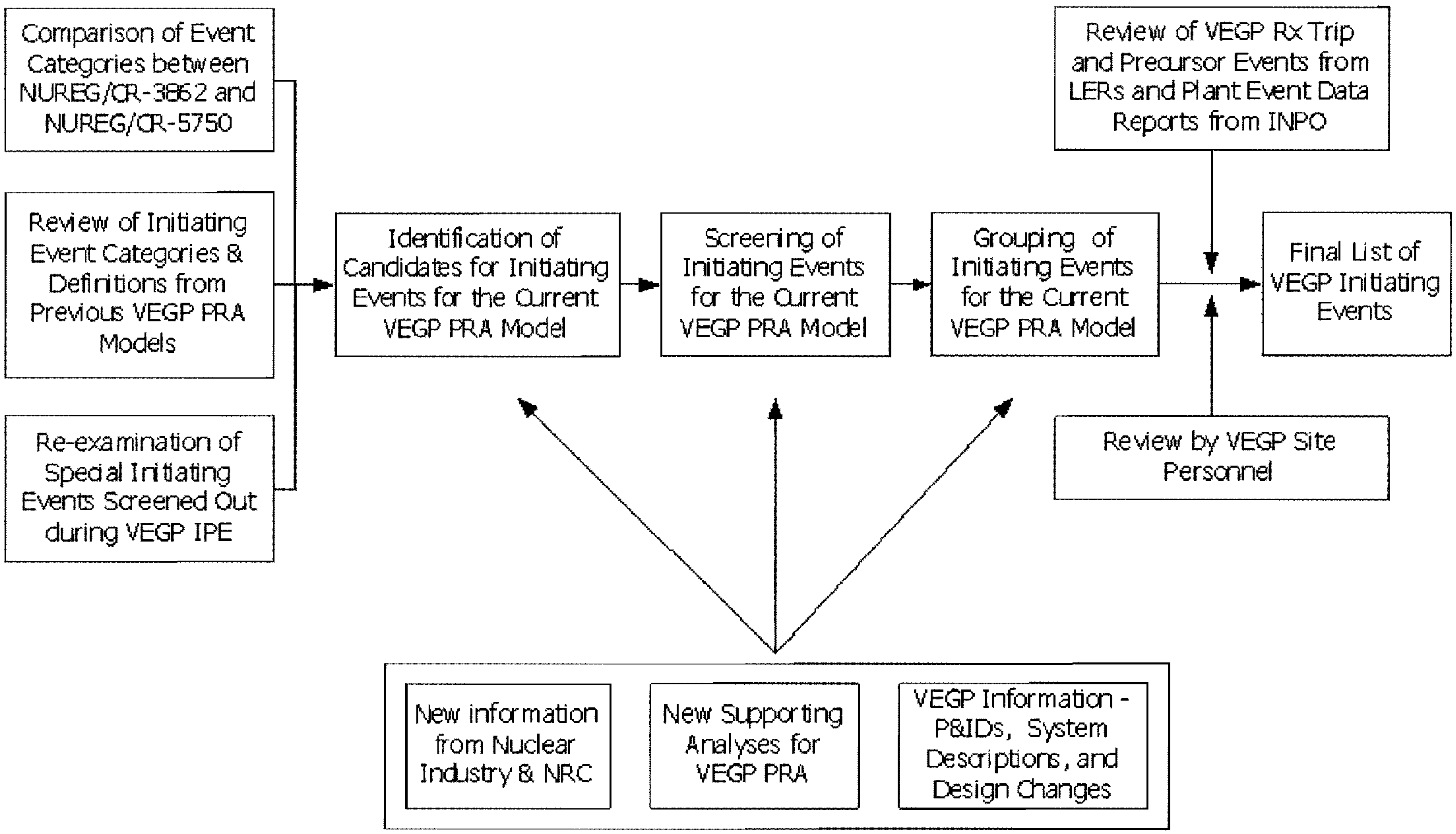


Figure 2.1-1 Initiating Event Update Tasks for the current VEGP PRA Model

The Initiating Events from the previous versions of the VEGP internal events PRA models including VEGP IPE were compared with the generic list of initiating events from the recent references, NUREG/CR-3862, and NUREG/CR-5750 in order to develop a revised initial list of Initiating Events. Also, a review of failure effects of all supporting systems was performed again in order to identify a list of special initiating events. During the review of supporting system failures, an added focus was given to re-examination of the supporting system failures, which were screened out in the IPE. Any event that causes an automatic reactor trip was considered as an initiating event. In addition, any event which requires a manual reactor trip within 8 hours was also considered as an initiating event. The 8 hour criteria comes from the average time for controlled plant shutdown. As a result of re-examination, loss of Class 1E 4.16 KV bus A and Loss of Class 1E 4.16KV Bus B were added to the current VEGP internal event PRA as new special initiating events.

Once candidate Initiating Events were identified, screening and grouping of these Initiating Events were performed. The grouping and screening were based on new generic information, results from VEGP-specific Supporting analyses, and the current VEGP design and operations.

Before finalizing the initiating events groups, VEGP plant specific events that had occurred during the period from 1998 to 2008 were reviewed to determine if there were reactor trip events or precursor events that could result in any unique initiating events that might not have been identified previously. The review did not find any additional unique initiating events.

Based on the above reviews, the list of all potential Initiating Events to be considered in the PRA was compiled. As a final check, the site VEGP personnel reviewed the screening and grouping of the Initiating Events to make sure any other important initiating events were not left out. A three-member site review team concluded that all the current VEGP initiating events were appropriate. In addition, the review team had a brainstorming session and suggested a list of initiating events (total of 44 events) that should be considered as initiating events for VEGP PRA. The review of these suggested initiating events revealed that all of them were already accounted for in one of the defined events on the VEGP Initiating Event list.

NRC RAI #13

F&O IF-C2a-01 states that "successful mitigation of ALL flood events is assumed to occur 30 minutes into any flood scenario." (emphasis in original) The response to this F&O states that the VEGP internal events flooding analysis does not credit operator actions for flood isolation/mitigation (that is, screening HEP values used were equal to 1.0) and that "screening human error probability (HEP) values in human induced flooding events do not make use of the results of the design-related calculations, which assume a 30 minute flow termination time." (emphasis added) The response to this F&O appears only to address human-induced flooding. Please clarify whether it is applicable to other sources of internal flooding such as pipe rupture.

SNC Response

The flooding scenario frequency of two scenarios initiated by pipe rupture was reduced by a factor of 0.1 (screening value) by crediting a human action. The remaining flooding scenarios initiated by a pipe rupture assumed no credit for flooding isolation.

The following provides additional information related to two scenarios initiated by pipe rupture for which credit for human action was taken.

The total internal flood induced CDF (LERF) contribution for Vogtle is ~7.9E-09 (9.34E-11). The flooding scenario frequency of two scenarios initiated by pipe rupture were reduced by a factor of 0.1 (screening value) by crediting a human action of failure to start a standby charging pump after a flooding induced loss of normal charging in the CVCS CHARGING PUMP PD ROOM. The flooding sources are pipe ruptures of:

(1) ACCW pipe size of 1-1/2" diameter/35 feet long and
(2) CVCS Discharge pipe size of 3"/ 30 feet long.

The two flooding scenarios have the following impact:

- Scenario 1:
  - Scenario frequency = 1.28E-07 (1.28E-06 X 0.1).
  - The conditional CDF (LERF) = 3.98E-07 (1.26E-09)
  - The total CDF (LERF) = 5.10E-14 (1.61E-16).
- Scenario 2:
  - Scenario frequency = 3.06E-06 (3.06E-05 X 0.1)
  - The conditional CDF (LERF) = 3.98E-07 (1.26E-09).
  - The total CDF (LERF) = 1.22E-12 (3.85E-15)

NRC RAI #14

F&O SY-B3-01 states that some systems "may be lacking common cause failure (CCF) grouping" but that these systems are non-risk significant and therefore would not impact categorization results. Assigning a CCF factor can substantively raise the failure likelihood and therefore the importance of a system. Please describe the criteria used to classify these systems as non-risk-significant and the basis (e.g. sensitivity study) for concluding that they would remain non-risk-significant had CCF been accounted for. Please discuss the interaction between the lack of CCF factors and the sensitivity studies whereby the CCF factors are increased and decreased.

SNC Response

A systematic analysis of common cause failure probability was performed for the revision 3.0 of the VEGP Internal Events (including Internal Flooding) PRA model. The results of this analysis are documented with the revision 3.0 of the model. When the model was revised to revision 4.0, the common cause analysis was reviewed and was found to be acceptable.

F&O SY-B3-01 is related to (and limited to) not modeling common cause failure to run for some non-safety related supporting system pumps such as Turbine Plant Cooling Water (TPCW) pumps. During normal operation, one TPCW pump is running while the other one is in standby. At no time both TPCW pumps are required to run during normal operation, which was the original basis for not modeling the common cause failure of TPCW pumps to run. Because a cause that fails a running TPCW pump could also affect the operation of the standby pump when it is demanded cannot be completely excluded, the common cause failure of TPCW pumps to run may be added to the model. The situation is similar to the common cause failure of Turbine Plant Closed Cooling Water (TPCCW) pumps to run. However, not modeling the common cause failure of TPCW and TPCCW pumps to run has minimal impacts on the results of VEGP PRA model because of the following reasons:

1. The largest contribution from the common cause failure of TPCW pumps or TPCCW pumps to VEGP risk would be the case when such a failure occurs during normal plant operation resulting in a reactor trip or initiating event because 1 year mission time should be considered in calculating the probability of such common cause failure event. However, not modeling such common cause failures of TPCW or TPCCW pumps in system fault tree models will not have any impact on plant risk because their contributions to the VEGP risk have already been captured as a part of the frequencies of initiating events like loss of feedwater, which is caused by loss of TPCW or TPCCW.

2. Common cause failure of TPCW or TPCCW pumps to run during 24 hours mission time after an initiating event occurs would have minimal impact on VEGP risk because of the following two reasons:

   a. In a configuration where one pump is normally running and another pump is in standby, failure of both pumps to run due to a common cause is less likely than the case where both pump are running all the time.

   b. A common cause failure of the Main Feedwater pumps is modeled in the PRA. The impact on the plant from loss of TPCW or TPCCW pumps would be the same as losing both Main Feedwater pumps. A review of cutsets (baseline model, revision 4.0) indicates that common cause basic event for the Main Feedwater pumps do not show up in the first 96,000 cutsets (1.1E-12). Therefore, an engineering judgment was exercised to not model the common cause event for TPCW or TPCCW pumps.

   The following two sensitivity runs were performed using the revision 4.0 of the VEGP Internal Events (including Internal Flooding) PRA model

   - Set all common cause events to 5^{th} percentile
   - Set all common cause events to 95^{th} percentile

   The review of cutsets obtained after setting all common cause basic events to their 5^{th} percentile value indicated that a common cause failure of the Main Feedwater pumps basic event did not show up at all in the 94,350 cutsets. The CDF value of the last cutset (#94,350) was 1.08E-15 per year.

The review of cutsets obtained after setting all common cause basic events to their 95th percentile value indicated that a common cause failure of the Main Feedwater pumps basic event showed up the first time in the 105,820th cutset. The corresponding CDF for this cutset was 1.52E-12 per year.

Based on the above information, it is evident that not modeling common cause event for the TPCW or TPCCW pumps will not have an impact on the categorization of SSCs per 10 CFR 50.69.

NRC RAI #15

F&O PP-A1. The "finding description" in the LAR Table 8 indicated that the peer review team identified some potentially significant fire areas that were determined to be outside the scope of the global analysis boundary, but no specific locations were included in Table 8. SNC's response states that none of the specific locations indentified in the F&O were screened. This implies that (1) the peer review team identified specific locations that were not included in Table 8 and (2) the peer review team misidentified them as missing (or screened out) but they were actually included in the fire PRA. Please identify the specific locations and clarify how these locations were dispositioned in the fire PRA.

SNC Response

As stated in the Resolution column of Table 8 in the LAR for F&O PP-A1-01, select "...locations were originally screened from the analysis". Thus, the peer review comment correctly identified that some structures within the Protected Area were screened without providing adequate justification. In order to resolve this F&O, a walk down was performed on April 6, 2012, to confirm the plant partitioning task for plant locations that were screened without adequate justification. The resolution to F&O PP-A1-02 is specific to the plant locations that were identified as missing from the documentation but were ultimately included in the model. A table summarizing the discrepancies in the identification of plant locations and resolutions follows. The associated model and documentation has been revised to reflect these changes. In addition, the criteria for qualitative screening is now clearly defined and discussed in the task report.

As stated in response to RAI #4, a focused peer review was performed on May 10, 2012. The revised documentation was reviewed as part of the focus scope peer review. The results of the focused peer review are summarized in RAI #4. As stated in the RAI #4, the focused peer review team considered F&Os related to element Plant Partitioning (PP) in their scope because it was an input to fire element QLS. Those locations that were screened were addressed in the focused peer review for the QLS element; the focused peer review team found all associated SRs as MET with no FINDINGS.

Discrepancies in Plant Locations			
Location	Peer Review Discrepancy	Resolution	PAU
Clean Lube Oil Storage Tank, Dirty Lube Oil Storage Tank and Auxiliary Boiler Fuel Oil Storage Tank (10)	Screened without sufficient justification	Included in the model	YARD
Hyperbolic Cooling Towers and Cooling Tower Canals and Basins (36) (One Each Per Unit)	Screened without sufficient justification	Included in the model	YARD
Maintenance Building (46)	Screened without sufficient justification	Included in the model	1530
High Voltage Switchyard Switch House (50)	Screened without sufficient justification	Included in the model	AHVSWYD
Low Voltage Switchyard Oil Collection Sump (51)	Screened without sufficient justification	Included in the model	ALVSWYD
High Voltage Switchyard Oil Collection Sump (52)	Screened without sufficient justification	Included in the model	AHVSWYD
Chemical and Electrical Equipment Building (53)	Screened without sufficient justification	Included in the model	1530
Ammonia Storage Tank (56) and Hydrazine Storage Tank (57)	Screened without sufficient justification	Included in the model	YARD

Discrepancies in Plant Locations			
Location	Peer Review Discrepancy	Resolution	PAU
Low Voltage Switchyard Valve Houses (58)	Screened without sufficient justification	Included in the model	ALVSWYD
High Voltage Switchyard Valve Houses (59, 65, 66)	Screened without sufficient justification	Included in the model	AHVSWYD
Nuclear Service Cooling Water (NSCW) Chemical Control Buildings (61) (One Per Unit)	Screened without sufficient justification	Included in the model	1532
High Voltage Switchyard Fence (68)	Screened without sufficient justification	Included in the model	AHVSWYD
Storage Buildings (99, 100)	Screened without sufficient justification	Included in the model	AHVSWYD
Radwaste Processing Facility (101)	Incorrectly identified as screened but included in the model.	Updated documentation	A350-RP
Outage Storage Building (103)	Missing from Documentation	Updated documentation and screened from analysis	N/A
Cable Storage Building (104)	Missing from Documentation	Updated documentation and screened from analysis	N/A
Tunnels on DWG AX4DJ8041 R.3, -8025 R.10 and -8046 R.4 (Artificial bldg no. T1)	Missing from Documentation but included in the model	Updated documentation	Multiple

Discrepancies in Plant Locations			
Location	Peer Review Discrepancy	Resolution	PAU
Tunnels on DWG AX4DJ8021 R.4 (Artificial bldg no. T2)	Missing from Documentation but included in the model	Updated documentation	Multiple
Tunnels on DWG AX4DJ8040 R.6 (Artificial bldg no. T3)	Missing from Documentation but included in the model	Updated documentation	Multiple
ALVSWYD (Building same)	Missing from Documentation but included in the model	Updated documentation	ALVSWYD
AHVSWYD (Building same)	Missing from Documentation but included in the model	Updated documentation	AHVSWYD
Pull Boxes in YARD (PB1)	Missing from Documentation and not included in the model	Updated documentation and included in the model	YARD
Pull Boxes in Low Voltage Switchyard (PB2)	Missing from Documentation and not included in the model	Updated documentation and included in the model	ALVSWYD
Pull Boxes in High Voltage Switchyard (PB3)	Missing from Documentation and not included in the model	Updated documentation and included in the model	AHVSWYD

NRC RAI #16

F&O. FSS-A1-01 questions the justification for screening out some ignition sources. The finding reported that SNC provided an example of screening criteria as being "sources were not used given potential for fire spread." Further support for screening these sources was that they are postulated to "have no consequential impact beyond itself (loss of only the fire source.)" F&O FSS-D3-01 provides a related observation that the ignition of secondary combustibles appeared limited. Please describe how SNC's evaluation considers the spread of a fire from the ignition source to other combustibles and how suppression activities are included in this evaluation.

SNC Response

The treatment of fire spread to secondary combustibles considers the potential for a larger zone of influence and the potential effect on hot gas layer because of the additional heat release rate from secondary combustibles. The Plant Vogtle fire scenario report describes the treatment of fire spread to secondary combustibles. The report identifies that the 98% heat release rate critical separation distances have been used to identify target sets for each ignition source. Additionally, secondary combustibles were identified for the ignition source. The potential for additional targets outside the zone of influence of an ignition source was evaluated based on NUREG/CR-6850 guidance for cable flame spread and tray propagation. NUREG/CR-6850 Section R.4.1 guidance was used to evaluate flame spread within a cable tray. Horizontal tray propagation was evaluated consistent with NUREG/CR-6850 Section R.4.2. The fire was postulated to propagate horizontally outwards at a 35 degree angle. NUREG/CR-6850 does not provide similar guidance for fire spread to adjacent vertical risers. Therefore, the fire PRA assumed that flame spread to adjacent vertical risers would occur similar to that described for adjacent horizontal trays in NUREG/CR-6850 Section R.4.2.2. Given the increase in heat release rate because of the potential for flame spread and tray propagation, additional targets from the increase in the zone of influence and hot gas layer effects were evaluated.

The Plant Vogtle fire scenario report also describes the treatment of suppression activities in the model. Manual suppression is implicitly credited when the electric panel factor methodology was applied. That is, the electric panel factor includes the consideration that manual suppression may prevent a challenging fire. When the electric panel methodology is not applied, manual suppression is credited in preventing target damage based on the time to damage targets. Also, manual suppression is credited in preventing hot gas layer based on the time to hot gas layer. The manual nonsuppression probabilities applied are based on Section 14 of Supplement 1 to NUREG/CR-6850. While the methodology can incorporate automatic suppression, the fire PRA did not take credit for automatic suppression to prevent fire spread or hot gas layer.

NRC RAI #17

FSS-A5-01 stated that transient ignition sources did not appear to be postulated in all possible locations. SNC's response indicated that some new transient fire locations were added, but that the change in the risk was minimal because, "the consequences of the postulated transient fire are bounded by another existing fire event." This implies that all the pinch points are exposed to fixed ignition sources or that SNC did not place transients fires at pinch points.

Please summarize how transient locations were selected, and how this process is consistent with the process of locating transients under pinch points as described in NUREG/CR-6850 (Reference 9). Note that an inaccessible area is not the same as a location where fire is simply unlikely, even if highly improbable. Please include a discussion of placement of transient fires in the control room as commented on by the peer review team in F&O FSS-B2-01.

Hot work should also be assumed to occur in locations where hot work is a possibility, even if improbable (but not impossible), keeping in mind the same philosophy. Please summarize how hot work fires are located and a frequency assigned.

SNC Response

For clarification, F&O FSS-A5-01 resolution discussion is reflective of the new transient fires added to the PRA. The consequence of these new transient fires was bounded by existing fixed ignition sources. However, transient fires previously selected did include "pinch point" locations that are not exposed to fixed ignition sources.

Transient fires were postulated at locations consistent with the guidance of NUREG/CR-6850. That is, transient fires were postulated at locations where fire PRA targets would be postulated to be damaged by transient ignition sources. Transient fire locations were not limited to only "pinch points" as defined by NUREG/CR-6850. The transient fire locations were selected by performing plant walkdowns or review of plant raceway drawings for locations in which a walkdown was not performed. Area accessibility was not used as a criterion when selecting transient fires.

The resolution to F&O FSS B2-01 included a walkdown of the control room and the selection of transient fires. Transient fires in the control room were selected at locations with fire PRA targets consistent with the discussion above.

Similar to the selection of transient fires, hot work fires were postulated at locations where fire PRA targets would be postulated to be damaged by hot work activity. The frequency was assigned consistent with the transient fire frequency. That is, the total hot work frequency of the plant area was apportioned to the postulated hot work fire based on the postulated target damage.

NRC RAI #18

FSS-A5-02 noted that the sum of the ignition frequencies in some physical area units (PAUs) appeared to differ from that expected after dividing up the frequencies in Task 6. The resolution indicted this was addressed by resolving FSS-A1-01 for fixed and FSS-A5-01 for transient ignition sources. Does the sum of the ignition frequencies now match that expected from dividing up the frequencies in Task 6?

SNC Response

The resolution to F&O FSS-A1-01 included a detailed review of the treatment of fixed ignition sources in the model for correctness. The Plant Vogtle fire scenario report contains a review of the fixed ignition source fire ignition frequencies that concludes the Physical Analysis Unit (PAU) fixed ignition source fire ignition frequency is appropriately apportioned. The resolution to F&O FSS-A5-01 included a review of plant locations and the selection of transient fires. The result of the review was that additional transient fires were selected throughout the plant. Given the resolution to F&O FSS-A1-01 and FSS-A5-01, the sum of postulated fire ignition frequencies do match that expected when the Task 6 fire ignition frequencies are distributed.

NRC RAI #19

FSS-C4-01 included observations related to credit for suppression activities and severity factors. SNC's response stated that SNC's method is consistent with a method that industry had reviewed (the unreviewed analysis method). Please describe the method used. If the method has not been accepted by the NRC, please provide a sensitivity study replacing the unacceptable method with the acceptable method indicating how great an impact this assumption has on the number of SSCs that would have been assigned low safety significant (LSS) but would now be HSS. This should include the standard categorization sensitivity studies (e.g., CCF and HEP evaluations) to the extent that the results of those sensitivity studies could also change.

SNC Response

To be provided within 120 days of the date of this letter

NRC RAI #20

FSS-C7-01. The peer review team identified missing dependencies between suppression activities in the multi compartment analyses and the hot gas layer scenarios. The response confirmed that some dependencies were not included, but also stated these were only found (and corrected) in the multi compartment analyses. What was the resolution of the peer review findings related to the hot gas layer scenarios?

SNC Response

The Plant Vogtle fire scenario report provides the discussion of the dependency between suppression systems. As stated in the report, automatic suppression systems were not credited in preventing hot gas layer. That is, only manual suppression is credited in the hot gas layer fire scenarios. The fire scenarios were reviewed and the credit for only manual suppression was confirmed. F&O FSS-C7-01 correctly described that only manual suppression was credited for hot gas layer fire scenarios. Therefore, dependency between suppression systems is not applicable to the hot gas layer scenarios and the resolution to F&O FSS-C7-01 was not applicable to the hot gas layer fire scenarios.

NRC RAI #21

FSS-D4-01. The peer review team stated, in part, that "...the heat release rate for transient fires in a number of (Physical Analysis Units) PAUs is assumed to be 69kW [kilowatts], which appears to be developed from an unreviewed analysis method (no specific reference to reviewed industry documents for this value is provided)." SNC's response justified the assumption by stating that "The overall treatment was consistent with the latest industry guidance as developed by an [Electric Power Research Institute] EPRI sponsored review effort and distributed to industry." The SNC response does not specifically identify the method (was it an unreviewed analysis method (UAM) nor state whether it was previously approved by the NRC staff, so the staff presumes that it was not previously approved. Please provide a sensitivity study that replaces the non-accepted method with a method that has previously been reviewed and approved by the NRC staff, indicating how great an impact this assumption has on the number of SSCs that would change from a low safety significant (LSS) category to a high safety significant category due to this change. This should include the standard categorization sensitivity studies (e.g., CCF and HEP evaluations) to the extent that the results of those sensitivity studies could also change.

SNC Response

At the time of the peer review, there had not been closure on the clarification for transient fires, which includes transient fire heat release rates. Since that time, there has been closure, and, in a letter dated June 21, 2012, (ML12172A406), the NRC did endorse the use of the method with minor clarifications for understanding. Therefore, the use of a lower heat release rate for transient fire in VEGP fire PRA was not based on an unendorsed method. Because the NRC endorsed method was used, SNC believes that a sensitivity analysis on this method is not necessary.

The VEGP fire PRA scenario report provides a discussion of the ranges of transient heat release rates presented in NUREG/CR-6850. Based on these ranges, two transient fire heat release rates were considered representative of the plant locations based on the location configuration. The VEGP fire PRA scenario report documents the heat release rate used in each plant location.

NRC RAI #22

FSS-E3-01, UNC-A2-01, UNC-A2-02. The peer review team noted that parameter uncertainty was not propagated through the fire PRA. SNC responded that all parameters that can be propagated with SNC tools have been propagated, and that conservative assumptions yielded conservative results so uncertainty analyses are not needed. Neither justification addresses the potential effects of uncertainty on the final safety-significance categories for SSCs and therefore are not sufficient to justify the use of CC I instead of CC II. Please meet the SR at CC II or justify the use of CC I for the categorization process.

SNC Response

SNC acknowledges the importance of understanding the impact of uncertainties on the categorization results. Table 3 of Enclosure 1 to NL-12-0932 provides an assessment of the various sources of uncertainty in the Vogtle fire PRA model and their potential impact on the categorization process. The assessment generally concludes that the fire PRA results are pessimistic (i.e., overestimates the fire CDF) due to conservative assumptions in the model. The issues identified in the fire PRA peer review F&Os noted in this question relate to estimation of parameter uncertainty.

The software tools available for calculating basic event importances do this on the basis of the point estimate representation of the mean. Thus, while performing a full propagation of the mean provides insight into the degree to which the calculated importances might be overestimated or underestimated, it would not be directly useful in the importance calculations without also accounting for the conservative bias introduced by the modeling uncertainties discussed in Table 3 of Enclosure 1 to NL-12-0932. While those modeling issues can be generally categorized as introducing a conservative bias, it is not possible to characterize the magnitude of this bias numerically, and so it is not meaningful to propagate the impacts through a parametric uncertainty analysis and attempt to apply this to SSC importances.

Since that qualitative assessment of sources of uncertainty indicates that the fire PRA results are pessimistic (i.e., overestimates the fire CDF due to conservative assumptions in the model), it is possible that the higher quantified fire PRA CDF could have some effect on the relative importance of some SSCs. Such potential impacts are addressed by the following:

- The fire PRA sensitivity studies defined in Section 5.2 of NEI-00-04 (and Section 5.12 of NMP-ES-065-001) provide a method to identify components that might become more risk significant.

- The process that SNC utilizes under our response to RAI #27 to replace Unendorsed Analysis Methods (UAMs) with endorsed methods will inherently eliminate the possible biasing of the fire PRA results that may have occurred as the result of having UAMs.

In summary, adequate consideration has been given to parametric uncertainty in the fire PRA, and this is not expected to affect the proposed application. Any potential conservative bias of the fire PRA results that could possibly arise due to 'methods' used, and potentially affect insights/results for this application, is addressed and eliminated by the use of NRC endorsed methods.

NRC RAI #23

FSS-G4-01. The peer review found no justification for crediting non-rated or active barriers. SNC's response simply stated that additional assessment was performed. What credit was taken for non-rated and active barriers and how was that credit developed?

SNC Response

NUREG/CR-6850 Section 11.5.4.4 guidance was used when applying credit to non-rated barriers. NUREG/CR-6850 guidance provides a screening barrier failure probability of 0.1, as well as failure probabilities in Table 11-3 for three barrier types. The barrier types in Table 11-3 were interpreted to not include non-rated barriers. In absence of additional guidance, only the NUREG/CR-6850 screening value of 0.1 was applied to non-rated barriers. Plant Vogtle fire scenario report identifies that the NUREG/CR-6850 value of 0.1 was used for non-rated barriers.

NUREG/CR-6850 Section 11.5.4.4 guidance was used when applying credit to active barriers. Fire doors and fire dampers are the two types of active barriers credited. NUREG/CR-6850 Table 11-3 values were used for fire door and fire dampers. The Plant Vogtle fire scenario report identifies that the NUREG/CR-6850 Table 11-3 values were used for these barriers.

NRC RAI #24

FQ-B1-04. The peer review team noted that the probabilities of consequently failed basic events were set to 1.0 instead of set to logical TRUE. SNC's response concluded that the difference had only a minor impact on total core damage frequency (CDF)/large early release frequency (LERF). Please evaluate the potential for this simplification to affect the importance measures and therefore the safety significance of SSCs.

SNC Response

The PRA model did set the failure basic events with a value of 1.0 to logical TRUE for quantification consistent with the ASME/ANS RA-Sa-2009 Standard supporting requirement.

SNC noticed that the resolution verbiage mentioned in the Table 8 of 50.69 LAR did not accurately characterize the situation. To clarify, the resolution discussion for F&O FQ-B1-04 is related to sequence flag events in the PRA logic model, which were assigned a 1.0 probability in the VEGP fire PRA logic model. These flags were added in order to provide additional information to the analysts during the cutset reviews in the model refinement process. These flags do not represent actual failures of SSCs. In quantifying VEGP fire PRA model, these flags were also set to logical TRUE to generate correct results. This is the baseline configuration; therefore, there is no issue related to simplification affecting the importance measures and the safety significance of SSCs.

NRC RAI #25

UNC-A2-01 noted that ignition frequencies from Section 10 of NUREG/CR-6850 were used. Supplement 1 states that a sensitivity analysis should be performed when using the fire ignition frequencies in the Supplement instead of the fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850. Provide the sensitivity analysis of the impact on using the Supplement 1 frequencies instead of the Table 6-1 frequencies on the importance measures and therefore the safety significance of SSCs for all of those bins that are characterized by an alpha that is less than or equal to one.

SNC Response

To be provided within 120 days of the date of this letter

NRC RAI #26

It was recently stated at the industry fire forum that the Phenomena Identification and Ranking Table Panel being conducted for the circuit failure tests from the DESIREE-FIRE and CAROL-FIRE tests may be eliminating the credit for Control Power Transformers (CPTs) (about a factor 2 reduction) currently allowed by Tables 10-1 and 10-3 of NUREG/CR-6850, Vol. 2, as being invalid when estimating circuit failure probabilities. Please perform a sensitivity study to quantify the impact of CPT credit on SSC categorization.

SNC Response

To be provided within 120 days of the date of this letter

NRC RAI #27

Please identify and provide technical justification for any fire PRA methodology that has not been formally accepted by the NRC staff. The NRC staff has formally accepted methods during resolution of UAMs as well as NUREG/CR-6850 (as supplemented) or the National Fire Protection Association Standard 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," frequently asked question guidance. Please evaluate the significance of the use of any method not yet accepted by the NRC Staff on the categorization process. If a position on a method has been established by the NRC, please confirm that the accepted version of the method is used per the NRC position and, if not, then provide a revised analysis and results using an accepted approach.

SNC Response

To be provided within 120 days of the date of this letter

Vogtle Electric Generating Plant
Request to Revise the Licensing Basis to Implement 10 CFR 50.69
Response to Request for Additional Information
Regarding Pilot 10 CFR 50.69 License Amendment Request

Enclosure 2

Requested Procedures

Procedure Owner: Amir Afzali / Risk-Informed Engineering Department Director / Corporate

(Print: Name / Title / Site)

Approved By: Owen Scott for Amir Afzali / 11/23/11

(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:	11/23/2011	N/A	N/A	11/23/2011	N/A
	Corporate	FNP	HNP	VEGP 1-2	VEGP 3-4

PRB Required

This NMP is under the oversight of the Risk-Informed Engineering Department

Writer(s):

Vish Patel

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Version Number	Version Description
1.0	Initial Issue

Table of Contents

1.0 Purpose

1.1 This procedure provides an overview of the process for implementing 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components [SSCs] for Nuclear Power Reactors.

1.1.1 The intent of 10 CFR 50.69 is to provide a means for appropriately focusing attention on those SSCs that are most important to safety, while maintaining reasonable confidence that other SSCs will be capable of performing their design basis functions.

1.1.2 To achieve this, 10 CFR 50.69 permits relaxation of the special treatment (controls) specified in certain other sections of the regulations for those SSCs that can be categorized as low safety significant.

1.2 This procedure is supplemented by the following detailed instructions/procedures that, together, form an integrated process for the categorization of SSCs.

- NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
- NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization
- NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components
- NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program
- NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

1.3 The process described in this procedure and the above-listed procedures/instructions satisfies the requirements of 10 CFR 50.69 (c), *SSC Categorization Process*, (d), *Alternative Treatment Requirements*, (e), *Feedback and Process Adjustment*, and (f), *Program Documentation, Change Control, and Records*, and (g), *Reporting*.

1.4 The process described in this procedure and the above-listed procedures/instructions is consistent with the following industry guidelines.

1.4.1 Nuclear Energy Institute (NEI) industry guidance document, NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline,* Revision 0.

1.4.2 Electric Power Research Institute (EPRI) Technical Report 1011234, 10 CFR 50.69 Implementation Guidance for Treatment of Structures, Systems and Components, Revision 0

2.0 Applicability

2.1 Categorization - This procedure is applicable only to those plant systems that have been selected for categorization. Since 10 CFR 50.69 is a voluntary rule, each site may decide which plant systems to categorize or not categorize. However, once a system is selected for categorization, <u>ALL</u> the components in that system <u>MUST</u> be included in the categorization process.

2.2 Treatment - The alternative treatment requirements allowed by 10 CFR 50.69 are available for use on low risk, safety related SSCs in categorized systems. The implementation of alternative treatment options is performed in a systematic and cost-effective manner that is Program-based (e.g., EQ program alternative requirements). Until alternative treatment requirements for a particular program are implemented through program and/or procedure changes, the previous requirements continue to apply.

3.0 Definitions

3.1 **Accident Sequence –** a representation in terms of an initiating event followed by a sequence of failures or successes of events (such as system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g. core damage or large early release).

3.2 **Alternative Treatment Requirements** – Those Owner-defined requirements applied to SSCs that provide reasonable confidence that 1) RISC-3 SSCs are capable of performing their safety related functions under design basis conditions; and 2) RISC-1 and RISC-2 SSCs perform their functions consistent with the key assumptions in the categorization process that relate to their assumed performance, as applicable. Alternative treatment requirements are differentiated from special treatment requirements in the use of "reasonable confidence" versus "reasonable assurance"

3.3 **Basic Safety Function (or Key Safety Function)** – one of the key safety functions of the plant, namely reactivity control, reactor pressure control, reactor coolant inventory control, decay heat removal, and containment integrity (It is noted that loss of a single train would typically not constitute a loss of a function). Note that the basic safety function is also known as key safety function.

3.4 **Completion Time (CT)** – the amount of time allowed for completing a required action. In the context of this case, the required action is to restore operability (as defined in the technical specifications, as applicable) to the affected system or equipment train.

3.5 **Complicated Initiating Event** – an event that trips the plant and causes an impact on a key safety function. Examples of complicated initiating events include loss of all feedwater (PWR/BWR), loss of condenser (BWRs).

3.6 **Conditional Consequence** – an estimate of an undesired consequence, such as core damage or a breach of containment, assuming failure of an item (e.g., conditional core damage probability (CCDP)).

3.7 **Conditional Core Damage Probability (CCDP)** – an estimate of the probability of an undesired consequence of core damage given a specific failure (e.g., piping segment failure).

3.8 **Conditional Large Early Release Probability (CLERP)** – an estimate of the probability of an undesired consequence of large early release given a specific failure (e.g., piping segment failure).

3.9 **Containment Barrier** – a component(s) that provides a containment boundary/isolation function including normally closed valves or valves that are designed to go closed upon actuation.

3.10 **Core Damage** – uncovering and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects. Such release would warrant implementation of off-site emergency response and protective actions.

3.11 **Core Damage Frequency (CDF)** - expected number of core damage events per unit of time.

3.12 **Defense-In-Depth** - the application of deterministic design and operational features that compensate for events that have a high degree of uncertainty with significant consequences to public health and safety. In accordance with Regulatory Guide 1.174, the defense-in-depth philosophy is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

3.13 **Failure –** as it applies to passive components, an event involving leakage, rupture, or other condition that would prevent an item from performing its intended safety function.

3.14 **Failure Mode –** as it applies to active components, a specific functional manifestation of a failure (i.e., the means by which an observer can determine that a failure has occurred) by precluding the successful operation of a piece of equipment, a component, or a system (e.g., fails to start, fails to run)

3.15 **Failure Modes and Effects Analysis (FMEA)** – a process for identifying failure modes of specific items and evaluating their effects on other components, subsystems, and systems.

3.16 **Failure Potential** – as it applies to passive components, likelihood of ruptures or leakage that result in a reduction or loss of the pressure-retaining capability of the item or the likelihood of a condition that would prevent an item from performing its safety function (e.g., fails to start, fails to run).

3.17 **High Safety Significant (HSS)** - those SSCs that are significant contributors to safety as identified through a blended risk-informed process that combines PRA insights, operating experience, and other technical information using IDP evaluations. This term is synonymous with the term “Safety Significant”.

3.18 **High Safety Significant Function (SSC)** – Same as Safety Significant Function (SSC).

3.19 **Initiating Event** - an event that perturbs the steady state operation of the plant by challenging plant control and safety systems whose failure could potentially lead to core damage and/or radioactive release. These events include human-caused perturbations and failure of equipment from either internal plant causes (such as hardware faults, floods, or fires) or external plant causes (such as earthquakes or high winds). Initiating events trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release.

3.20 **Integrated Decision-Making Panel (IDP)** – a multi-discipline panel of plant-knowledgeable experts that reviews the results of the initial categorization of SSCs/functions to ensure that the appropriate considerations from plant design and operating practices and experience are reflected in the categorization input. For the purpose of supporting the categorization effort detailed herein, the needed expertise on the IDP <u>SHALL</u> include PRA, safety analysis, plant operations, design engineering, and system engineering.

3.21 **Large Early Release** – the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects.

3.22 **Large Early Release Frequency (LERF)** - expected number of large early releases (releases of airborne fission products from containment) per unit of time.

3.23 **Low Safety Significant (LSS)** - those SSCs that are not significant contributors to safety as identified through a blended risk-informed process that combines PRA insights, operating experience, and other technical information using IDP evaluations.

3.24 **Low Safety Significant Function (SSC)** – a function (SSC) for which the Integrated Decision-Making Panel has applied a risk-informed process that combines PRA insights, operating experience, and other technical information to determine that safety significance is not high.

3.25 **Non-Modeled Hazards** – Any of the following risk hazards for which there does not exist an approved PRA quantification model:

- Fire risk
- Seismic risk
- Other External risks (e.g., high winds, external floods)
- Shutdown risk

3.26 **Operator Recovery Action** – a human action performed to regain equipment or system operability from a specific failure or human error in order to mitigate or reduce the consequences (e.g., loss of a system, loss of a pump train, indirect effects) of the failure.

3.27 **Passive Component** – pressure retaining components and active components with a pressure retaining function.

3.28 **Piping Segment** – a portion of piping, components, or a combination thereof, and their supports, in which a failure at any location results in the same consequence (e.g., loss of a system, loss of a pump train, indirect effects)

3.29 **Plant Mitigative Features** – systems, structures, and components that can be relied on to prevent an accident or that can be used to mitigate the consequences (e.g., loss of a system, loss of a pump train, indirect effects) of an accident

3.30 **Pressure-Boundary Failure** - piping segment failures involving ruptures or leakage that result in a reduction or loss of the item's pressure-retaining capability.

3.31 **Probabilistic Risk Assessment (PRA)** - a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public.

3.32 **Qualitative Insights** - an assessment of the safety significance of an SSC based on the collective judgment of IDP members and utilizing a systematic process that supplements the PRA results.

3.33 **Risk Informed Safety Classification (RISC)** – a method outlined in 10 CFR 50.69 for classifying SSCs into one of the following categories:

- RISC-1: SSCs that are safety-related and perform safety-significant functions.
- RISC-2: SSCs that are non-safety-related and perform safety-significant functions.
- RISC-3: SSCs that are safety-related and perform low safety-significant functions.
- RISC-4: SSCs that are non-safety-related and perform low safety-significant functions.

3.34 **Risk Metrics** – a determination of what activity or conditions produce the risk, and what individual, group, or property is affected by the risk.

3.35 **Safety Related –** Plant structures, systems, and components necessary to assure:

- The integrity of the reactor coolant pressure boundary,
- The capability to shut down the reactor and maintain it in a safe shutdown condition, or
- The capability to prevent or mitigate the consequences of accidents, which could result in off-site exposures that exceed the guidelines established in 10CFR100.

3.36 **Safety Significance** - the relative importance of an SSC in protecting the reactor core and/or preventing a negative impact on the health and safety of the public.

3.37 **Safety Significant** - those SSCs that are significant contributors to safety as identified through a blended risk-informed process that combines PRA insights, operating experience, and other technical information using IDP evaluations. This term is synonymous with High Safety Significant (HSS).

3.38 **Safety-significant function (SSC)** - a function (SSC) whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk. Determination of safety significance is made by the Integrated Decision-Making Panel using a risk-informed process that combines PRA insights, operating experience, and other technical information. [Note: loss of a single train would typically not constitute a loss of a function]

3.39 **Sensitivity Studies** - analyses that are performed to ensure that assumptions or uncertainties made in the PRA are not masking the importance of an SSC. Typical sensitivity studies include increasing and decreasing human error rates, increasing and decreasing common cause failure rates, increasing and decreasing maintenance unavailability, and increasing the failure rate of LSS components. Sensitivity studies can also be used to address issues raised during the IDP process and may include other bounding quantitative assessments designed to demonstrate that an SSC is not safety significant.

3.40 **Special Treatment Requirements** - NRC requirements imposed on SSCs that go beyond normal industry-established (industrial) controls and measures and are intended to provide reasonable assurance that the equipment is capable of meeting its design bases functional requirements under design basis conditions. These additional special treatment requirements include, for example, design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements.

3.41 **Spatial Effect** – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, harsh environment, debris generation or flooding.

3.42 **Success Criteria** – criteria for establishing the minimum number or combination of systems or components required to operate, or minimum levels of performance per component during a specific period of time, to ensure that the safety functions are satisfied.

3.43 **Train** – As used in this procedure/instruction, a train consists of a set of equipment (e.g., pump, piping, associated valves, motor, and control power) that individually fulfills a safety function (e.g., high-pressure safety injection) with a mean unavailability of 1E-02 as credited in Tables 2 and 3 of NMP-ES-065-002. A half train (0.5 trains) SHALL have a mean unavailability of 1E-01, 1.5 trains SHALL have a mean unavailability of 1E-03, etc.

3.44 **Treatment** - Activities, processes, and/or controls that are performed or used in the design, installation, maintenance, and operation of SSCs as a means of 1) Specifying and procuring SSCs that satisfy performance requirements; 2) Verifying over time that performance is maintained; 3) Controlling activities that could impact performance; and 4) Providing assessment and feedback of results to adjust activities as needed to meet desired outcomes.

3.45 **Treatment Program** – That program which implements the special treatment requirements that have been identified in 10 CFR 50.69 as no longer being required for low safety significant SSCs. Examples of treatment programs include the Maintenance Rule and the Equipment Qualification Program.

3.46 **Unaffected Backup Train** – for passive component assessment, a train that is not adversely impacted (i.e., failed or degraded) by the postulated piping failure in the FMEA evaluation. Impacts can be caused by direct or indirect effects of the postulated piping failure.

4.0 Responsibilities

4.1 Director, Risk-Informed Engineering

- Manages the 10 CFR 50.69 Program
- Ensures PRA technical adequacy as required to support the 10 CFR 50.69 process
- Assigns Risk-Informed Application Engineer(s) as required to support the Program
- Provides training to IDP members and other selected site personnel

4.2 Site IDP

- Evaluates PRA risk insights, passive risk insights, and deterministic risk insights to reach a consensus-based categorization for system functions and components.
- Reviews results from performance monitoring and periodic reassessments to ensure that the basis for the categorization of SSCs remains valid and that any implemented alternative treatments have not significantly degraded the performance of the associated components.
- Evaluates recommended changes to categorization results due to PRA model updates, changes to the plant, changes to operational practices, as well as other applicable changes.

4.3 Cognizant Risk-Informed Application Engineer

- Provides PRA insights in support of the active risk categorization of system functions and components.
- Provides PRA insights in support of the passive risk categorization of system components.
- Provides the results of other hazards analyses for those hazards that are not modeled in the PRA.

4.4 Cognizant System Engineer

- Develops system functions.
- Maps each component in the system to the system function(s) supported.
- Participates in the categorization of active risk for system functions and components.
- Participates in the categorization of passive risk for system components.

4.5 Operations Representative

- Provides deterministic responses to the essential questions used to assess the risk of system functions.
- Participates in the categorization of active risk for system functions and components.
- Participates in the categorization of passive risk for system components

4.6 Treatment Program Owner (for each Program)

- Evaluates alternative treatment options for RISC-3 SSCs
- Evaluates whether additional controls are necessary for RISC-2 SSCS
- Evaluates whether additional controls are necessary for RISC-1 SSCs to ensure acceptable performance for beyond design basis functions.
- Implements alternative treatment requirements or other changes as identified above.

4.7 Director, Nuclear Licensing

- Updates the Final Safety Analysis Report, following the implementation of 10 CFR 50.69, to reflect which systems have been categorized (Requirement from 10 CFR 50.69, part f.2)
- Submits a licensee event report for any event or condition that would have prevented RISC-1 or RISC-2 SSCs from performing a safety-significant function (Requirement from 10 CFR 50.69, part g).

5.0 Procedure

NOTES

- This procedure has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Activities described in this procedure may be performed prior to NRC approval of the license amendment.
- The alternative treatment requirements specified in 10 CFR 50.69 (d) SHALL NOT be implemented UNLESS step 5.1, step 5.2, and, if applicable, step 5.3 are verified to be complete.

5.1 After the license amendment is approved by the NRC, **perform** a documented evaluation to ensure that the process described in this procedure meets the requirements of, and is consistent with, the NRC-approved license amendment.

5.2 **Track** the performance of this evaluation via a Condition Report action. This evaluation SHALL be approved by the Director, Risk-Informed Engineering and by the Director, Licensing. After approval of the evaluation, **revise** this procedure to remove the above Note, this step and steps 5.1 and 5.3.

5.3 IF the above evaluation concludes that the process described in this procedure DOES NOT meet the requirements of, or is inconsistent with, the approved license amendment, THEN **revise** this procedure accordingly and **re-perform** any evaluations or activities already performed using the revised procedural requirements.

NOTES

- This procedure provides a summary of the processes for 1) categorizing SSCs and 2) applying alternative treatment requirements to RISC-3 SSCs.
- Step 5.4 SHALL NOT be implemented until the license has been amended to incorporate 10 CFR 50.69.

5.4 The Nuclear Licensing (NL) Department SHALL **update** the Final Safety Analysis Report when treatments are implemented. The NL Department SHALL **submit** a licensee event report for any event or condition that would have prevented RISC-1 and RISC-2 SSCs from performing a safety significant function.

5.5 **Refer** to Attachment 1, Figure 1 for a summary description of the relationship of this procedure (NMP-ES-065) with associated instructions and NMP-ES-066 (General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program).

5.6 **Ensure** that the following requirements have been satisfied prior to using this procedure.

5.6.1 Training

Specific training and qualifications requirements for IDP members and designated alternates are detailed in NMP-ES-066-002.

Other individuals who may participate in the IDP meetings, such as the cognizant system engineer for the system under discussion, should be generally familiar with the categorization process.

5.6.2 PRA Capability

Ensure that the PRA is appropriately detailed and of sound technical quality in accordance with the following considerations:

1) At a minimum, the PRA MUST model severe accident scenarios resulting from internal initiating events occurring at full power operation.
2) Importance measures related to core damage frequency (CDF) and large early release frequency (LERF) are used to identify safety significant SSCs.
3) Other risk contributors are also assessed either by PRA modeling or by bounding analyses or screening assessments. These other risk hazards are fire risks, seismic risks, other external risks (e.g., tornados, external floods, etc), and shutdown risks.
4) Sensitivity studies are performed for LSS PRA-modeled components to ensure sufficient margins exist.

5.7 **Refer** to the following to gain an understanding of the Categorization Process.

> **NOTE**
>
> Detailed guidance on the Categorization Process is provided in References 7.4, 7.5, and 7.6.

5.7.1 Risk Categories

SSCs <u>SHALL</u> be categorized as RISC-1, RISC-2, RISC-3, or RISC-4.

5.7.2 Blended Risk Approach

The categorization process blends PRA risk insights with deterministic insights to arrive at a consensus-based risk category for system functions and components. In addition, the risk of passive components or the passive function of active components is separately determined through a similar PRA-deterministic process. The final risk of components is the higher of the PRA risk, deterministic risk, or passive risk (if applicable).

5.7.3 Qualitative Insights

> **NOTE**
>
> Qualitative insights should be used to supplement the PRA risk results. Due to PRA assumptions and limitations, such as those mentioned above, qualitative insights are typically needed to categorize components within a particular plant system, primarily because many components in a particular system are not modeled by the PRA. In addition, these insights can provide an alternate and valuable perspective that can be blended with the PRA results to reach an overall risk assessment.

Qualitative insights include, but are not necessarily limited, to the following:

- Supplementary analyses that are used to compensate for PRA limitations in quantifying the risk during plant shutdown and for hazards that may not modeled such as fire risks, seismic risks, and other external risks (e.g., tornadoes, external floods, etc.)
- Qualitative risk assessment that considers, like the PRA, the impact and likelihood of failure of the SSC under consideration.
- Plant design bases
- Maintenance of defense-in-depth
- Maintenance of sufficient safety margins
- Plant and industry operating experience
- Operational and maintenance processes

5.7.4 Passive (Pressure Retention) Risk of Components

Components having only a pressure retaining function (also referred to as passive components), and the passive function of active components are required to undergo a separate process in order to determine their passive risk, in accordance with the following:

- This process is based on the EPRI risk-informed in-service inspection (RI-ISI) evaluation, supplemented by additional qualitative considerations.
- Each piping component (including valves and supports) is categorized as HSS or LSS based on the consequence evaluations of an assumed pressure boundary failure.
- The consequence evaluations use both PRA and qualitative insights.

5.7.5 Overall Categorization

SSCs that are considered HSS based on PRA results, deterministic results, or evaluation of passive risk (if applicable), SHALL be categorized as RISC-1 or RISC-2. Otherwise, they can be categorized as RISC-3 or RISC-4.

5.7.6 Integrated Decision Making Panel

SSC categorization SHALL be performed by an IDP, staffed with expert, plant-knowledgeable members. For the purpose of the categorization process, the expertise of the IDP members SHALL include, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering. The IDP evaluates PRA risk results along with deterministic insights and defense-in-depth to arrive at consensus-based categorization decisions.

5.7.7 Risk Significant Attributes

For each HSS component, **identify** the attributes of the component that are associated with its safety significance.

5.7.8 Scope of SSC categorization

The categorization process is a voluntary process that may be applied to selected plant systems or structures. However, ONCE a system selection is made, THEN ALL the components within the system or structure are to be categorized, NOT just specific components within a system or structure. The categorization scope for a particular system or structure includes ALL system or structure components associated with that system AND possessing a unique component identification number in the Plant Data Management System (PDMS).

5.7.9 Periodic Reviews and Performance Feedback

For those SSCs that have been categorized, periodic reviews SHALL be conducted to ensure continued validity of categorization results and to review SSC performance. Changes to plant design, operational practices, and industry and plant operational experience should be evaluated for impact on existing categorizations.

5.8 **Refer** to the following to gain an understanding of the Application of Alternative Treatments (Ref. 10 CFR 50.69 (d), *Alternative Treatment Requirements*).

5.8.1 RISC-3 components are removed from the scope of the following special treatment requirements:

- Maintenance Rule [10 CFR 50.65]
- Environmental Qualification [10 CFR 50.49]
- Seismic Qualification [Portions of Appendix A to 10 CFR Part 100]
- Applicable portions of ASME XI repair & replacements, with limitations [10 CFR 50.55a(g)]
- Applicable Portions of IEEE standards [10 CFR 50.55a(h)]
- In-service Testing [10 CFR 50.55a(f)]
- In-service Inspection [10 CFR 50.55a(g)]
- Local Leak Rate Testing [10 CFR 50 Appendix J]
- Quality Requirements [10 CFR 50 Appendix B]
- Deficiency Reporting [10 CFR Part 21]
- Event Reporting [10 CFR 50.55(e)]
- Notification Requirements [10 CFR 50.72]

5.8.2 It is important to note that although the above requirements will no longer be applicable to RISC-3 components, 10 CFR 50.69 does not eliminate the design requirement that RISC-3 components be capable of performing their design basis functions. Rather, 10 CFR 50.69 provides for the use of alternative treatments to provide "reasonable confidence that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life."

5.8.3 Treatment Program procedures or guidelines that implement the above special treatment requirements should be revised to recognize that RISC-3 components are removed from the scope and to identify acceptable alternative treatments, as applicable, to provide reasonable confidence that these components would perform their design basis function.

5.8.4 UNTIL alternative treatment requirements for a particular program are implemented through program and/or procedure changes, **continue** to apply the previous requirements.

5.8.5 The general approach for modifying a typical special treatment program to incorporate RISC-3 components would involve the following activities:

- **Identify** Purpose and Scope of the Existing Programs
- **Identify** Basis of Existing Program and Special Treatment Requirements
- **Identify** Requirements that No Longer Apply
- **Identify** Alternate Treatment Elements that Support Design Basis
- **Develop** Alternate Treatment Options for RISC-3 Items

5.8.6 RISC-2 components SHALL be evaluated in order to determine if additional controls or treatments should be applied, considering their risk significance and operational performance.

5.8.7 RISC-1 components SHALL continue to be subject to existing special treatment requirements. However, in accordance with 10 CFR 50.69, RISC-1 components SHALL also be evaluated to determine if additional requirements are necessary to ensure that the performance of these components remains consistent with the assumed performance in the categorization process (including the PRA) for beyond design basis functions.

5.8.8 Other Considerations

The objective of implementing 10 CFR 50.69 is to allow increased focus and resources to be applied to safety significant SSCs. Given this, plant processes and procedures associated with the operation and maintenance of the plant should be revised to take advantage of the categorization results and the reduction of treatment requirements. The general approach is to increase focus and attention on RISC-1 and RISC-2 components while allowing increased flexibility for RISC-3 and RISC-4 components. Processes that would benefit from this approach include but are not limited to:

- Preventive Maintenance
- Corrective Maintenance
- Condition Reporting
- Design Change Control
- Procurement
- Work Control
- Quality Inspections

6.0 Records

This procedure itself does not generate records. However, instructions associated with this procedure generate records. Refer to these instructions for guidance on how records will be maintained.

7.0 References

7.1 10 CFR 50.69, Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors

7.2 NEI 00-04, 10 CFR 50.69 SSC Categorization Guide, Revision 0

7.3 EPRI Technical Report 1011234, 10 CFR 50.69 Implementation Guidance for Treatment of Structures, Systems and Components, Revision 0

7.4 NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights

7.5 NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization

7.6 NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components

7.7 NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

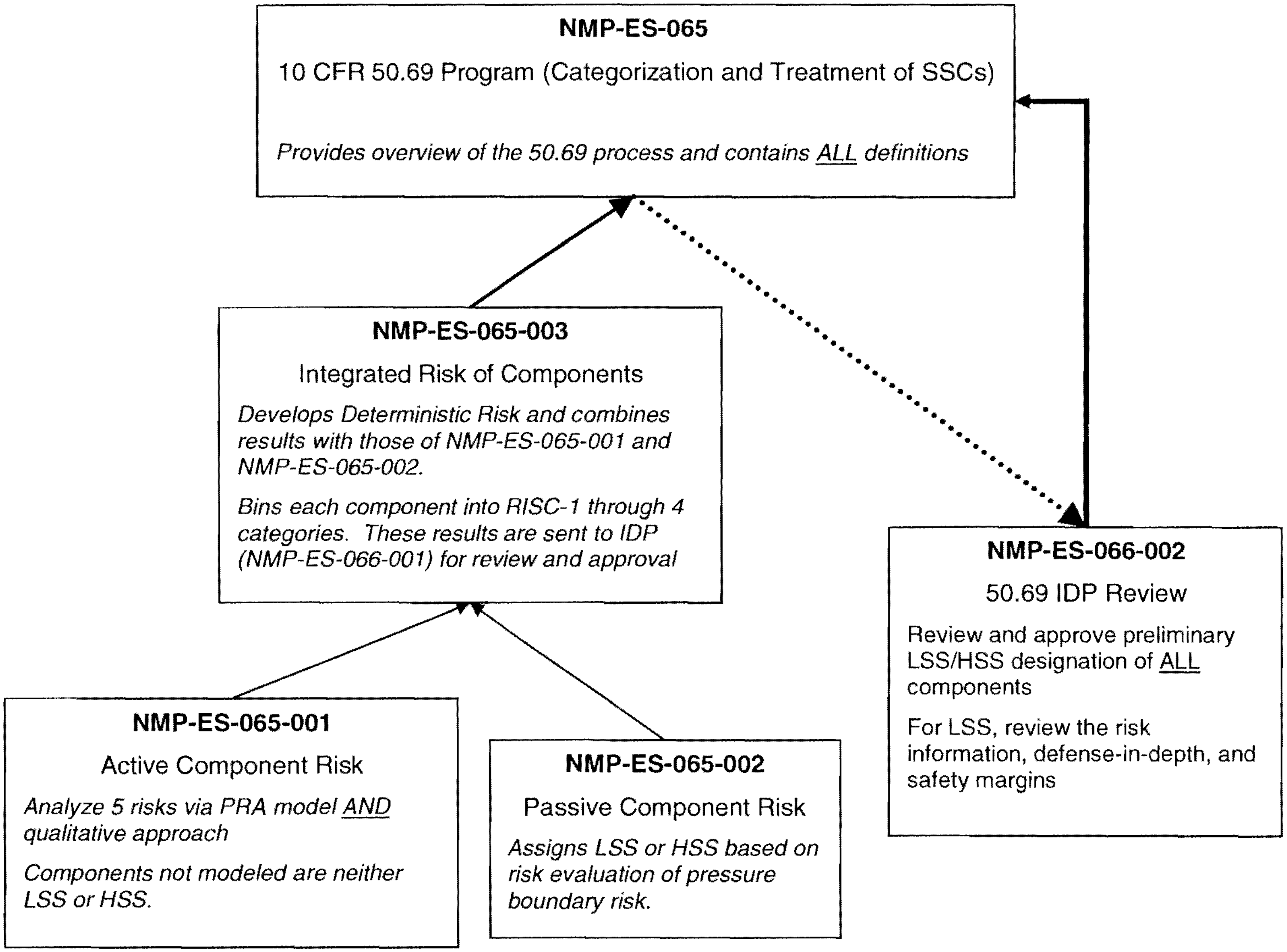
7.8 NMP-ES-066-002: Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

7.9 NMP-NL-XXX, Nuclear Licensing Procedure for Implementation of 10 CFR 50.69 (TBD)

8.0 Commitments

None

Attachment 1, Figure 1: Summary of relationship of this procedure (NMP-ES-065) with associated instructions and NMP-ES-066 (Integrated Decision-Making Panel General Guidance For Risk Informed SSC Categorization Program and Independent Decision-Making Panel For Surveillance Frequency Control Program)



Instruction Owner: Amir Afzali / Risk-Informed Engineering Department Director / Corporate
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Approved By: Owen Scott for Amir Afzali / 11/23/11
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:	11/23/2011	N/A	N/A	11/23/2011	N/A
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PRB Required

This NMP is under the oversight of the Risk-Informed Engineering Department

Writer(s):

Vish Patel

Plant Review Board (PRB) review and approval is required for this NMP

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Revision Description

Version Number	Revision Description
1.0	Initial issue

1.0 {tc "1.0 Purpose" \f C \l 1}Purpose

1.1 This instruction provides guidance to support the determination of risk significance of active structures, systems, and components (SSCs) in accordance with 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.*

1.2 This instruction is part of an integrated categorization process which includes the following additional procedures/instructions

- NMP-ES-065, 10 CFR 50.69 Program
- NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
- NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Structures, Systems, and Components
- NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

1.3 The process described in this instruction and the above-listed procedures/instructions is considered to satisfy the requirements of 10 CFR 50.69 (c), SSC Categorization Process, (e), Feedback and Process Adjustment, and (f), Program Documentation, Change Control, and Records. The scope of this instruction does NOT include alternative treatment requirements specified in 10 CFR 50.69 (d).

2.0 {tc "2.0 Applicability" \f C \l 1}Applicability

2.1 This instruction is applicable only to those plant systems that have been selected for categorization. Since 10 CFR 50.69 is a voluntary rule, each Site may decide which plant systems to categorize or not categorize. However, once a system is selected for categorization, ALL the components in that system MUST be included in the categorization process.

2.2 This instruction is applicable to activities involving the 10 CFR 50.69 Active Component Categorization performed by Southern Nuclear Operating Company (SNC) personnel or supplemental personnel.

3.0 {tc "3.0 Definitions" \f C \l 1}Definitions

All definitions are contained in NMP-ES-065. This instruction SHALL be used with NMP-ES-065.

4.0 {tc "4.0 Responsibilities" \f C \l 1}Responsibilities

> **NOTE**
>
> General responsibilities for the 10 CFR 50.69 Process, including the activities described in this instruction, are found in NMP-ES-065. Specific responsibilities unique to this instruction are described as follows.

4.1 Cognizant Risk-Informed Application Engineer

- Provides the internal events at power PRA base case risk importances for SSCs in the system under review and for system SSCs modeled in the PRA...
- Provides the results of other hazards analyses risk importances and insights for SSCs in the system under review for those hazards that are NOT modeled in the PRA.
- Provides the results of the integrated risk importance analysis for SSCs in the system under review.
- Provides the results of sensitivity studies of the impact of uncertainties in assumptions, such as those related to common cause, human reliability, and failure rates for SSCs that are candidate LSS.
- Provides additional PRA Model insights which may influence the SSC categorization outcome.
- Provides PRA risk changes resulting from model updates or other factors that could impact existing SSC categorizations.
- Participates in the periodic performance review process and analyzes the impact of changes in performance of SSCs categorized as LSS on the risk significance results

4.2 Cognizant System Engineer

- Provides the list of systems, functions, and associated SSCs for which risk significance information is required.
- Provides design basis and severe accident functions of SSCs relative to each hazard evaluated.

5.0 {tc "5.0 Procedure" \f C \l 1}Procedure

NOTES

- This procedure has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Activities described in this procedure may be performed prior to NRC approval of the license amendment.
- The alternative treatment requirements specified in 10 CFR 50.69 (d) SHALL NOT be implemented UNLESS step 5.1, step 5.2, and, if applicable, step 5.3 are verified to be complete.

5.1 {tc "5.1 through 5.3 - License Amendment Review" \f C \l 2}After the license amendment is approved by the NRC, **perform** a documented evaluation to ensure that the process described in this procedure meets the requirements of, AND is consistent with, the NRC-approved license amendment.

5.2 **Track** the performance of this evaluation via a Condition Report action. This evaluation SHALL be approved by the Director, Risk-Informed Engineering AND by the Director, Licensing. After approval of the evaluation, **revise** this procedure to remove the above Note, this step and steps 5.1 and 5.3.

5.3 IF the above evaluation concludes that the process described in this procedure DOES NOT meet the requirements of, or is inconsistent with, the approved license amendment, THEN **revise** this procedure accordingly and **re-perform** any evaluations or activities already performed using the revised procedural requirements.

NOTE

Appropriate steps in the following process are to be documented, including the basis. As applicable, this documentation should be entered into a database and coded where practical in order to facilitate data manipulation and retrieval tasks.

5.4 {tc "5.4 Risk Categories " \f C \l 2}Risk Categories

For the risk hazards identified herein, **categorize** SSCs as HSS or LSS in accordance with this instruction.

5.5 {tc "5.5 PRA Capability " \f C \l 2}PRA Capability

5.5.1 **Ensure** that the PRA is appropriately detailed and of sound technical quality in accordance with the following considerations:

1) At a minimum, the PRA must model severe accident scenarios resulting from internal initiating events occurring at full power operation.
2) NRC expectations for PRA capability for 10 CFR 50.69 categorization application are that the internal events at power PRA will have been peer reviewed against the requirements in the ASME/ANS PRA Standard (i.e., RA-Sa-2009 - Ref. 7.11 - or subsequent revisions) as endorsed with NRC clarifications in Regulatory Guide 1.200 (Ref. 7.10), AND shown to meet most requirements in that standard at capability category II or better.
3) PRA limitations may include hazards that are NOT modeled (e.g., external initiating events), plant shutdown risks, AND SSCs that are NOT modeled. These limitations can be addressed through supplementary analyses. Typically, these involve bounding analyses or qualitative methods such as screening assessments and/or IDP evaluations.

5.5.2 IF there are areas where the PRA does NOT meet a requirement at capability category II, THEN **perform** a documented assessment regarding the potential impact of such limitations on the 10 CFR 50.69 categorization application and the manner in which they will be compensated for in using the PRA. **Perform** such an assessment for each PRA model used in the categorization process (e.g., internal events at power, internal fire, seismic, etc.).

5.5.3 IF any components in the system being categorized are modeled in the PRA, THEN **communicate** to the IDP the following information as a basis for the adequacy of the PRA risk results used in the categorization process:

- A characterization of the adequacy of the PRA
- PRA limitations, including but not necessarily limited to hazards that are not modeled (e.g., external initiating events), plant shutdown risks, AND SSCs that are not modeled.

5.6 {tc "5.6 Use of PRA for SSC Importance Determination " \f C \l 2}Use of PRA for SSC Importance Determination

5.6.1 **Assess** the importance of an SSC by identifying the PRA basic events that represent the SSC. The types of basic events are:

- Events that explicitly model the performance of an SSC (e.g., pump X fails to start);
- Events that implicitly model an SSC (e.g., some human actions, initiating events, etc.); OR
- A combination of both types of events

5.6.2 **Identify** the events in the PRA that can be used to represent each SSC. Within this mapping, **record** whether the PRA: 1) explicitly models the performance of the SSC (e.g., pump X fails to start); 2) implicitly models SSC (e.g., via assumption for availability to support a human action, as a contributor to an initiating event, etc.); OR 3) uses a combination of both types of events

5.6.3 **Address** the contribution of common cause to a component's importance. IF a component DOES NOT have a common cause basic event in the PRA to be included in the computation of importances, THEN **assess** whether a common cause event should be added to the model.

5.7 {tc "5.7 Risk Characterization Overview " \f C \l 2}Risk Characterization Overview [per NEI-00-04, Ref. 7.2]

5.7.1 **Assess** the following risk hazards separately:

NOTE

Separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

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

5.7.2 IF there are multiple PRAs (e.g., a PRA for internal events and a PRA for fire risks), THEN **perform** an integrated assessment of the modeled risk hazards to reach an overall conclusion about the risk significance of each SSC from a PRA perspective.

5.7.3 **Utilize** the following guidance from NEI 00-04 to determine acceptable methods for assessing each of the risk hazards:

Table 5-1
Summary of Risk Significance Characterization Used in NEI 00-04

Risk Hazard	Acceptable Approaches	Scope of Safety-Significant SSCs
Internal Events	PRA Required	Per PRA Risk Ranking
	Screening Approaches NOT Allowed	n/a
Fire	Fire PRA	Per PRA Risk Ranking
	FIVE (Fire Induced Vulnerability Evaluation)	All SSCs Necessary to Maintain Low Risk (See 5.13)
Seismic	Seismic PRA	Per PRA Risk Ranking
	SMA (Seismic Margins Analysis)	All SSCs Necessary to Maintain Low Risk (see 5.16)
High Winds, External Floods, etc.	PRA	Per PRA Risk Ranking
	IPEEE Screening	All SSCs Necessary to Protect Against Hazard (see 5.18)
Shutdown	Shutdown PRA	Per PRA Risk Ranking
	Shutdown Safety Plan	All SSCs Required to Support Shutdown Safety Plan (5.20)

5.8 {tc "5.8 Risk Evaluations – General Requirements" \f C \l 2}Risk Evaluations based on PRA or Other Hazards Analyses – General Requirements

Provide the following risk assessment results for the risk hazards identified in Table 5-1:

- For components that are modeled by one or more PRAs, an integrated importance assessment (per Section 5.21) of LSS or HSS for each such component.
- For any of the above hazards that are NOT modeled in the PRA, the results of the hazards evaluations (bounding, qualitative, or screening) that indicate which components are considered HSS.
- For modeled components that are identified as having an integrated importance assessment of LSS, the results of the required sensitivity studies.
- Modeled components that are identified as having an integrated importance assessment of LSS AND are within 10% of the threshold for HSS (referred to as buffer zone components).

5.9 {tc "5.9 Internal Events PRA Risk Importance " \f C \l 2}Internal Events at Power Risk Importance Using the Internal Events at Power PRA

5.9.1 **Utilize** the internal events at-power PRA to quantify the risk importance measures for the identified functions AND SSCs in the system of interest as described in this section and as depicted in Attachment 1.

5.9.2 **Apply** this risk importance process, including sensitivity studies for both CDF and LERF.

5.9.3 **Utilize** the following Table 5-2 to determine the risk significance of the components.

Table 5-2 Risk Importance Criteria for HSS
Sum of F-V for all basic events modeling the SSC of interest, including common cause events, > 0.005
Maximum of component basic event RAW values > 2
Maximum of applicable common cause basic events RAW values > 20

5.9.4 IF the component risk importance results for both CDF and LERF are below ALL of the above thresholds, THEN **identify** the component as candidate LSS. Otherwise, **identify** the component as candidate HSS.

NOTES

1) In calculating the F-V risk importance measure, it is recommended that a CDF (or LERF) truncation level of five orders of magnitude below the baseline CDF (or LERF) value be used for linked fault tree PRAs. In addition, the truncation level used should be sufficient to identify all functions with RAW>2.

2) In cases where the internal events CDF (or LERF) is dominated by an internal flooding result that has a conservative bias, it is appropriate to break the evaluation of importance measures into two steps. This prevents the conservative bias of the flooding analysis from masking the importance of SSCs NOT involved in flood scenarios.

   - The first step uses importance measures computed using the entire internal events PRA.
   - The second step uses importance measures computed without the dominant contributor included. This prevents "masking" of importance by the dominant contributor.

5.9.5 **Identify** the PRA basic events that represent the SSCs of interest.

5.9.6 **Create** a mapping of those components to be categorized to the events in the PRA that can be used to represent each component, in accordance with the following:

- Within this mapping, **record** whether the PRA explicitly models the performance of the component (e.g., pump X fails to start), implicitly models the component (e.g., via assumption for availability to support a human action, as a contributor to an initiating event, etc.), or treats the component as a combination of both types of events.
- IF a component of interest does NOT have a common cause event in the PRA to be included in the computation of importances, THEN **assess** whether a common cause event should be added to the model.

5.9.7 **Determine** the answer to the following question: Does the PRA model importance quantification process accounts for the contribution of the component's role in initiating events? That is, if a component is a contributor to a complicated initiating event (e.g., loss of NSCW or loss of CCW for PWRs, loss of condenser for BWRs), does the PRA model that initiator contribution explicitly (i.e., within the fault tree model) such that the component importances reflect both the mitigation AND initiating event contribution?

- IF the answer is YES, THEN **conclude** that the PRA importance measures provide sufficient scope to perform the initial screening AND **perform** Steps 5.9.8 through 5.9.10 to determine the component's candidate safety significance.
- IF the answer is NO, THEN **perform** Steps 5.9.8 through 5.9.11 to determine the component's candidate safety significance

5.9.8 For each component of interest, **calculate** the F-V and RAW for that component, using the internal events at power PRA in accordance with the following:

- The F-V importance of a component is the sum of the F-V importances for the failure modes of the component relevant to the function being evaluated.

NOTE
Risk Reduction Worth (RRW) is also an acceptable measure in place of F-V because the F-V criteria can be readily converted to RRW criteria.

- The RAW importance of a component is the maximum of the RAW values computed for basic events involving failure modes of the individual component.
- The RAW importance of the common cause events involving a component must also be evaluated. The maximum of the applicable common cause basic event RAW values is used.

5.9.9 IF ANY of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate HSS AND **document** its safety significant attributes. **Use** Table 5-3 for guidance.

5.9.10 IF NONE of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate LSS AND **perform** sensitivity studies per Section 5.10.

5.9.11 For those components for which the PRA model importance quantification process does NOT account for the contribution of the component's role in initiating events, the following evaluations are required.

5.9.11.1 Determine the answer to the following question: Does the component exceed ANY of the risk importance criteria in Table 5-2?

- IF the answer is YES, THEN **classify** the component as candidate HSS. **Identify** complicated initiating events for which F-V importance is > 0.005 AND **determine** if the component can directly cause one of these complicated initiating events
  - o IF the component can directly cause a complicated initiating event with F-V > 0.005, THEN **document** the component's safety significant attributes relative to both mitigation AND event initiation.
  - o IF the component CANNOT directly cause a complicated initiating event with F-V > 0.005, then **document** the component's safety significant attributes relative only to mitigation.

- IF the answer is NO, AND the component can directly cause a complicated initiating event with F-V > 0.005, THEN **classify** the component as candidate HSS AND **document** the component's safety significant attributes relative to event initiation.
- IF the answer is NO, AND the component CANNOT directly cause a complicated initiating event with F-V > 0.005, THEN **classify** the component as candidate LSS AND **perform** sensitivity studies per Section 5.10.

Table 5-3 EXAMPLE IMPORTANCE SUMMARY (NEI-00-04 Table 5-1)

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

In this example, valve 'A' would be considered candidate safety significant on two bases, either one of which would be sufficient to identify the component as candidate safety-significant:
(1) The total F-V exceeded the criterion of 0.005, and
(2) The RAW criterion was also met for the common cause group including valve 'A'. Note that valve 'A', valve 'B' and valve 'C' would be identified as candidate safety-significant due to this criterion.
The component failure mode which contributes significantly to the importance of valve 'A' is failure to open (failure modes 1, 4, 5 and 6 as shown above). This failure mode is used in the identification of safety-significant attributes. If an individual failure mode had not alone exceeded the screening criteria, then the significantly contributing failure modes would be used in defining the attributes.

5.10 {tc "5.10 Internal Events Sensitivity Studies " \f C \l 2}Internal Events at Power PRA Sensitivity Studies

> **NOTE**
>
> Sensitivity studies are not needed for the system that is being categorized if the importance measures computed using the Internal Events at Power PRA indicate that ALL modeled components, including non-safety-related components, are HSS. However, IF some of the components are LSS, THEN sensitivity studies are used to determine whether other conditions might lead to the component being safety-significant, based on the same F-V and RAW criteria used in the base case

5.10.1 Perform the recommended sensitivity studies identified in Table 5-4 below.

Table 5-4 Sensitivity Studies For Internal Events PRA (adapted from NEI-00-04 Table 5-2)

Sensitivity Study
• Increase all human error basic events to their 95th percentile value
• Decrease all human error basic events to their 5th percentile value
• Increase all component common cause events to their 95th percentile value
• Decrease all component common cause events to their 5th percentile value
• Set all maintenance unavailability terms to 0.0
• Any applicable sensitivity studies identified in the characterization of PRA adequacy and identification of important assumptions and sources of uncertainty.

5.10.2 **Perform** the sensitivity studies on human error rates, common cause failures, AND maintenance unavailabilities to ensure that assumptions of the PRA are NOT masking the importance of an SSC. In these sensitivity studies, **make** the indicated changes to ALL of the associated basic events in the PRA, NOT just those associated with the system being categorized. For example, in the first sensitivity, the 95^{th} percentile values are used for ALL HEPs in the PRA.

5.10.3 In cases where plant-specific uncertainty distributions are NOT readily available, **review** other PRAs to identify appropriate parameter ranges.

> **NOTE**
>
> Experience with plant-specific PRAs has shown that the variations in distributions are relatively small, especially with respect the ratio of the mean and 95th percentile values in lognormal distributions (the most common distribution used in PRAs). Guidance on evaluation of uncertainty, and identification of important and key assumptions and sources of uncertainty in the PRA, is provided in EPRI TR-1016737.

5.10.4 IF the sensitivity studies identify that the component could be safety-significant, THEN **classify** the component as candidate HSS AND **identify** the safety-significant attributes that yielded that sensitivity studies conclusion.

5.10.5 IF the sensitivity studies identify that the component is NOT safety-significant, THEN **classify** it as candidate LSS from an internal events at power risk perspective. In this case, **identify** the qualitative reasons as to why the component is of low risk significance from the internal events at power perspective (e.g., does not perform an important function, there is excess redundancy in the system or function, low frequency of challenge, etc.).

5.10.6 IF one or more of the sensitivity studies identify the component as candidate HSS, THEN **document** this information, including the associated explanation, as part of the risk significance categorization information package to be presented to the IDP (Ref. 7.6 and 7.7).

5.11 {tc "5.11 Fire PRA Risk Importance " \f C \l 2}Internal Fire Risk Importance Evaluation using Fire PRA

> **NOTE**
>
> For plants with a fire PRA, the safety significance process is generally the same as the process for an internal events at power PRA. For the Fire PRA, the risk importance process used for the internal events at power PRA is slightly modified to consider the fact that most fire PRAs DO NOT have the ability to aggregate the mitigation importance of a component with the fire initiation contribution. For that reason, components are evaluated using standard importance measures for their mitigation capability only.

5.11.1 **Utilize** the Fire PRA to quantify the fire risk importance measures for the identified SSCs in the system of interest, as depicted in Attachment 2.

5.11.2 IF the fire PRA CDF, including ALL screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), THEN **classify** the SSCs modeled in the Fire PRA as LSS from a fire perspective and skip to Step 5.12.

Note

Fire suppression systems that are evaluated using the fire risk analysis can be categorized using this process. However, in order to apply this categorization process to suppression systems, specific sensitivity studies may be required to identify their relative importance, consistent with F-V and RAW (guarantee success/failure). In general, fire barriers would NOT be in the scope of this guideline unless the fire risk analysis allows the quantification of the impacts of failure of the barrier.

- In cases where the impact of fire barrier failure can be evaluated in the risk analysis, the categorization process is applicable.
- Sensitivity studies should be used to identify the role a barrier plays in maintaining risk levels.

5.11.3 **Apply** the Fire risk importance process to both CDF and LERF.

NOTE

Where LERF CANNOT be quantitatively linked into the fire model, the insights from the internal events LERF model should be qualitatively coupled with the assessment of fire impacts on containment isolation to develop recommendations for the IDP on LERF contributors.

5.11.4 For each component of interest, **use** the fire PRA to calculate the F-V and RAW for that component.

5.11.5 IF ANY of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate HSS AND **document** its safety significant attributes. **Use** Table 5-3 for guidance.

5.11.6 IF NONE of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate LSS AND **perform** sensitivity studies per Step 5.12.

5.12 {tc "5.12 Fire PRA Sensitivity Studiel" \f C \l 2}Internal Fire PRA Risk Importance Sensitivity Studies

> **NOTE**
>
> Sensitivity studies are not needed for the system that is being categorized if the importance measures computed using the Fire PRA indicate that ALL modeled components, including non-safety-related components, are HSS. However, if some of the components are LSS, then sensitivity studies are used to determine whether other conditions might lead to the component being safety-significant, based on the same F-V and RAW criteria used in the base case

5.12.1 Perform the recommended sensitivity studies identified in Table 5-5 below.

Table 5-5 Sensitivity Studies For Fire PRA (adapted from NEI-00-04 Table 5-3)

Sensitivity Study
• Increase all human error basic events to their 95th percentile value
• Decrease all human error basic events to their 5th percentile value
• Increase all component common cause events to their 95th percentile value
• Decrease all component common cause events to their 5th percentile value
• Set all maintenance unavailability terms to 0.0
• No credit for manual suppression
• Any applicable sensitivity studies identified in the characterization of PRA adequacy and identification of important assumptions and sources of uncertainty.

5.12.2 IF the sensitivity studies identify that the component could be safety-significant, THEN **classify** the component as candidate HSS from a fire risk perspective AND **identify** the safety-significant attributes that yielded that sensitivity studies conclusion.

5.12.3 IF the sensitivity studies identify that the component is NOT safety-significant, THEN **perform** the following additional checks:

- IF such a component is NOT safety related, THEN **classify** it as candidate LSS from a fire risk perspective.
- IF such a component IS safety-related, THEN **identify** the qualitative reasons as to why the component is of low fire risk significance (e.g., does not perform an important function, there is excess redundancy in the system or function, low frequency of challenge, etc.), AND **classify** the component is retained as candidate LSS from a fire risk perspective.

5.13 {tc "5.13 Fire Hazard Assessment Without Fire PRA " \f C \l 2}Internal Fire Safety Significance Without Fire PRA

NOTES

- For plants for which a fire PRA has not been developed, NEI-00-04 allows the use of the EPRI *Fire Induced Vulnerability Evaluation* (FIVE) methodology, which is a process to assist in identifying potential fire susceptibilities and vulnerabilities. As SNC plants do not have FIVE analyses, the alternative approach selected for plants without a fire PRA is to use the plant Fire Safe Shutdown analysis.
- Although this is a departure from NEI-00-04, it represents an additional deterministic conservatism in the process, as it will reduce the benefit that might otherwise be derived from a risk-informed categorization of fire risk importance using a fire PRA.

5.13.1 For each component, **identify** the fire design basis AND severe accident functions of the component.

5.13.2 **Review** the plant's Fire Safe Shutdown analysis to determine if the component is credited as part of the safe shutdown paths evaluated, in accordance with the following:

- IF a component is credited as part of a fire safe shutdown path, THEN **classify** it as HSS AND **identify** the attributes which yielded that conclusion. For example, **document** which key safety function(s) the component supports in the Fire Safe Shutdown analysis, AND any relevant assumptions in the Fire Safe Shutdown analysis regarding component availability or reliability.
- IF the component does NOT participate in the safe shutdown path, THEN **classify** it as candidate LSS.

5.14 {tc "5.14 Seismic PRA Risk Importance " \f C \l 2}Seismic Risk Importance Evaluation using Seismic PRA

NOTES

- For plants with a seismic PRA, the safety significance process is generally the same as the process for an internal events at power PRA.
- For the seismic PRA, the risk importance process used for the internal events at power PRA is slightly modified to consider the fact that seismic events cannot be caused by plant components, hence there is no initiation contribution to importance.
    - For that reason, components are evaluated using standard importance measures for their mitigation capability only.

5.14.1 **Utilize** the seismic PRA to quantify the seismic risk importance measures for the identified SSCs in the system of interest, as depicted in Attachment 2.

5.14.2 IF the seismic PRA CDF, including ALL screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), THEN **classify** the SSCs modeled in the seismic PRA as LSS from a seismic perspective and skip to Step 5.15.

NOTE

SSCs may have been screened out of the seismic PRA due to inherent seismic robustness. That is, in the development of the seismic PRA, certain SSCs may have been judged to have sufficiently high seismic capability that they would not be significant contributors to seismic risk within the capability of the seismic risk model, and therefore not included in the model. For such screened SSCs, regardless of their categorization outcome, it is important that the inherent seismic robustness that allows them to be screened out of the seismic PRA should be retained. For example, categorization of such screened components as LSS should not be viewed as implying that they do not need to retain their design seismic capability (they do). These considerations are necessary to maintain the validity of the categorization process.

5.14.3 **Apply** the seismic risk importance process to both CDF and LERF.

NOTE

Where LERF CANNOT 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 isolation to develop recommendations for the IDP on LERF contributors.

5.14.4 For each component of interest, **use** the seismic PRA to calculate the F-V and RAW for that component.

5.14.5 IF ANY of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate HSS AND **document** its safety significant attributes. **Use** Table 5-3 for guidance.

5.14.6 IF NONE of the risk importance criteria listed in Table 5-2 are exceeded for a component, THEN **identify** that component as candidate LSS AND **perform** sensitivity studies per Step 5.15.

5.15 {tc "5.15 Seismic PRA Sensitivity Studies " \f C \l 2}Seismic PRA Risk Importance Sensitivity Studies

NOTE

Sensitivity studies are not needed for the system that is being categorized if the importance measures computed using the seismic PRA indicate that ALL modeled components, including non-safety-related components, are HSS. However, if some of the components are LSS, then sensitivity studies are used to determine whether other conditions might lead to the component being safety-significant, based on the same F-V and RAW criteria used in the base case.

5.15.1 Perform the recommended sensitivity studies identified in Table 5-6 below.

Table 5-6 Sensitivity Studies For Seismic PRA (adapted from NEI-00-04 Table 5-4)
Sensitivity Study
• Increase all human error basic events to their 95th percentile value
• Decrease all human error basic events to their 5th percentile value
• Increase all component common cause events to their 95th percentile value
• Decrease all component common cause events to their 5th percentile value
• Set all maintenance unavailability terms to 0.0
• Use correlated fragilities for all SSCs in a given area
• Any applicable sensitivity studies identified in the characterization of PRA adequacy and identification of important assumptions and sources of uncertainty.

5.15.2 IF the sensitivity studies identify that the component could be safety-significant, THEN **classify** the component as candidate HSS from a seismic risk perspective AND **identify** the safety-significant attributes that yielded that sensitivity studies conclusion.

5.15.3 IF the sensitivity studies identify that the component is NOT safety-significant, THEN **perform** the following additional checks:

- IF such a component is NOT safety related, THEN **classify** it as candidate LSS from a seismic risk perspective.

- IF such a component IS safety-related, THEN **identify** the qualitative reasons as to why the component is of low seismic risk significance (e.g., does not perform an important function, there is excess redundancy in the system or function, low frequency of challenge, etc.), AND **classify** the component is retained as candidate LSS from a seismic risk perspective.

5.16 {tc "5.16 Seismic Risk Assessment Without Seismic PRA " \f C \l 2}Seismic Safety Significance Without Seismic PRA

> **NOTE**
>
> For plants for which a seismic PRA has not been developed, NEI-00-04 allows the use of the seismic margins methodology (e.g., as performed for the IPEEE), which 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.

5.16.1 For each component, **identify** the seismic design basis AND severe accident functions of the component.

5.16.2 **Review** the plant's Seismic Margins Analysis to determine if the component is credited as part of the safe shutdown paths evaluated.

- IF a component is credited as part of a seismic-margins-evaluated safe shutdown path, THEN **classify** it as HSS AND **identify** the attributes which yielded that conclusion.
- IF the component is NOT included in the seismic safe shutdown path, THEN **classify** it as candidate LSS.

5.17 {tc "5.17 External Hazards PRA Risk Importance " \f C \l 2}Other External Hazards Risk Evaluation Using PRA

> **NOTE**
>
> For plants with an External Hazards PRA, the safety significance process is generally the same as the process for an internal events at power PRA. For the External Hazards PRA, the risk importance process used for the internal events at power PRA is slightly modified to consider the fact that external events cannot be caused by plant components, hence there is no initiation contribution to importance. For that reason, components are evaluated using standard importance measures for their mitigation capability only.

5.17.1 **Determine** whether the system or structure is evaluated in the external hazards PRA. Personnel knowledgeable in the scope, level of detail, AND assumptions of the external hazards PRA should make these determinations.

5.17.2 IF a system/structure is NOT involved in an external hazards PRA, THEN **classify** the SSC as candidate LSS from the standpoint of other external risks and skip to Step 5.19.

5.17.3 IF the system or structure is determined to be evaluated in the external hazards PRA, THEN **continue** the evaluation steps in this section.

5.17.4 **Utilize** the External Hazards PRA to quantify the external hazards risk importance measures for the identified SSCs in the system of interest, as depicted in Attachment 2.

5.17.5 IF the External Hazards PRA CDF, including ALL screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), THEN **classify** the SSCs modeled in the External Hazards PRA as LSS from an External Hazards perspective and skip to Step 5.19.

5.17.6 **Apply** the Other External Hazards risk importance process to both CDF and LERF.

NOTE

Where LERF CANNOT be quantitatively linked into the other external events model, the insights from the internal events LERF model should be qualitatively coupled with the assessment of other external events impacts on containment isolation to develop recommendations for the IDP on LERF contributors.

5.17.7 Follow the evaluation process steps for seismic risk importance evaluation, in sections 5.14 and 5.15 (sensitivity studies as indicated in Table 5-6; note that the sensitivity for "correlated fragilities" applies AND should be interpreted as fragilities related to the other hazard in question.)

5.18 {tc "5.18 External Hazards Assessment Without PRA " \f C \l 2}Other External Hazards Risk Evaluation Without PRA

NOTE

For plants for which an External Hazards PRA has not been developed, NEI-00-04 allows the use of the external hazards screening evaluation performed to support the requirements of the IPEEE. Personnel knowledgeable in the scope, level of detail, and assumptions of the external hazards analysis should make these determinations.

5.18.1 IF the SSC is NOT involved in an external hazards screening evaluation, THEN **classify** the SSC as candidate LSS from the standpoint of other external risks.

5.18.2 IF the SSC is evaluated in the external hazards screening analysis, THEN **determine** the safety significance of the SSC in accordance with the following steps:

5.18.3 For each component, **identify** the other external hazard design basis AND severe accident functions of the component.

5.18.4 **Review** the plant's IPEEE other external hazards screening evaluation to determine if the component is credited as part of the safe shutdown paths evaluated.

- IF a component is credited as part of a an other external hazards-evaluated safe shutdown path, THEN **classify** it as HSS AND **identify** the attributes which yielded that conclusion.
- IF the component is NOT included in the other external hazards-evaluated safe shutdown path, THEN **classify** it as candidate LSS if one of the following conditions is TRUE:
  - The component either did NOT participate in any external hazard scenarios that were screened during the external hazards evaluation; or
  - Even if credit for the component was removed, the screened scenario would NOT become unscreened.

5.19 {tc "5.19 Shutdown PRA Risk Importance " \f C \l 2}Shutdown Safety Assessment Using Shutdown PRA

NOTE

For plants with a shutdown PRA that is comparable to an at-power PRA (i.e., generates annual average CDF/LERF), the safety significance process is generally the same as the process for an internal events at power PRA. This process is shown in Attachment 1.

5.19.1 **Follow** the process defined in steps 5.9 and 5.10 using the shutdown PRA.

5.19.2 IF the Shutdown PRA CDF, including ALL screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), THEN **classify** the SSCs modeled in the Shutdown PRA as LSS from a shutdown perspective and skip to Step 5.21

5.20 {tc "5.20 Shutdown Safety Assessment Without PRA " \f C \l 2}Shutdown Safety Assessment Without PRA (Using NUMARC 91-06 Program)

NOTES

1. NUMARC 91-06 specifies that a defense in depth approach should be used with respect to each defined shutdown key safety function. This is generally accomplished by designating a running and alternative system/train to accomplish the given key safety function. The shutdown safety assessment process guidance provided in NEI-00-04 is utilized in this Section to determine safety significance from a shutdown perspective.
2. In this assessment, primary shutdown safety system and first alternative shutdown safety system refer to a system or systems with the following attributes:
   - It has a technical basis for its ability to perform the function.
   - It has margin to fulfill the safety function.
   - It does not require extensive manual manipulation to fulfill its safety function.

5.20.1 IF multiple systems/trains are available to satisfy the key safety function, THEN **classify** those SSCs that support the primary AND first alternative methods to satisfy the key safety function as candidate HSS.

5.20.2 IF the SSC's failure would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.), THEN **classify** it as candidate HSS.

5.20.3 IF the SSC is NOT included in either of the manners identified in 5.20.1 or 5.20.2, THEN **classify** it as candidate LSS.

5.21 {tc "5.21 Integral Assessment of Overall Risk Significance " \f C \l 2}Integral Assessment of Overall Risk Significance

NOTES

- Each risk contributor is initially evaluated separately in the preceding steps in order to avoid reliance on a combined result that might mask the results of individual risk contributors, due to the significant differences in the methods, assumptions, conservatisms, and uncertainties associated with the risk evaluation of each.
    - In general, the quantification of risks due to external events and non-power operations tend to contain more conservatisms than internal events, at-power risks. As a result, performing the categorization simply on the basis of a mathematically combined total CDF/LERF would lead to inappropriate conclusions. For example, an SSC that is very important for a hazard that contributes only 1% to the total CDF/LERF may be found to have low importance measures when the integral assessment is performed. Therefore, it is desirable in a risk-informed process to understand safety significance from an overall perspective, especially for SSCs that were found to be safety-significant due to one or more of these risk contributors.
- Note that the integral risk assessment addresses <u>ALL</u> of the PRA-modeled SSCs, not only those that have already been determined to be safety significant. However, the integrated importance <u>CANNOT</u> be higher than the maximum of the individual measures.

5.21.1 **Compute** the CDF for the integrated importance measures by weighting 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, in accordance with the following formulas.

<u>Integrated F-V Importance:</u>

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

Where,

IFV_i = Integrated F-V Importance of Component i over all CDF Contributors (i.e., the set of contributors for which PRAs are available and used in the categorization, e.g., internal events, fire, seismic and shutdown)

$FV_{i,j}$ = F-V Importance of Component i for CDF Contributor j

CDF_j = CDF of Contributor j

<u>Integrated Risk Achievement Worth Importance:</u>

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

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

5.21.2 Once calculated, **perform** an assessment of these integrated values against the screening criteria of F-V >0.005, RAW > 2.0 for individual basic events, AND RAW > 20 for common cause basic events, using the following considerations.

- An SSC that is very important for a hazard that contributes only 1% to the total CDF/LERF would be found to have very low importance measures when the integrated assessment is performed.
- In no case should the importance from the integral assessment 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.

5.21.3 **Utilize** the same process for calculating LERF, if available.

5.21.4 **Document** the results of the integrated assessment for presentation to the IDP as part of the categorization input package. This integrated assessment allows the IDP to determine whether the safety significance of the SSC should be based on the significance for that individual hazard or from the overall integrated result, avoiding a strict reliance on a mathematical formula that ignores the significant dissimilarities in the calculated risk results.

5.22 {tc "5.22 Risk Sensitivity Study Background " \f C \l 2}Risk Sensitivity Study Background

Refer to the following to gain an understanding of the need for and the use of risk sensitivity studies for evaluation of the risk implications of changes in special treatments.

- An overall risk sensitivity study is required by the process defined in NEI-00-04. This sensitivity study should be performed for each individual plant system as the categorization of its functions is provided to the IDP. A sensitivity study should be performed for the system, and a cumulative sensitivity for ALL the SSCs categorized using this process. This is intended to provide the IDP with both the overall assessment of the potential risk implications and the relative contribution of each system.
- 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, based on the following considerations.
    - One of the guiding principles of this process is that changes in treatment should NOT significantly degrade performance for RISC-3 SSCs and should maintain or improve the performance of RISC-2 SSCs.
    - Thus, it is anticipated that there would be little, if any, net increase in risk.
    - This risk sensitivity study is made using the available PRAs to evaluate the potential impact on CDF and LERF, based on a postulated change in reliability.
    - For categorizations that rely on PRAs, this sensitivity is useful because the importance measures used in the initial safety significance assessment were based on the individual SSCs considered. Changes in performance can influence not only the importance measures for the SSCs that have changes in performance, but also others. Thus, the aggregate impact of the changes should be evaluated to assess whether new risk insights are revealed.

- It is NOT necessary to address the cumulative impact of SSCs for hazards where screening tools such as SMA were used because if they are included in the screening analysis they are considered HSS, thus there would be no change in treatment and no change in performance.
- Risk sensitivity studies should be realistic, i.e., should not model unreasonable increases in component unreliability. In this risk sensitivity study, the unreliability of ALL modeled low safety-significant SSCs is increased simultaneously by a common multiplier as an indication of the potential trend in CDF and LERF, if there were a degradation in the performance of low safety significant SSCs. A factor of between 3 and 5 is recommended in NEI-00-04. However, the particular factor value is determined specific to the plant, based on a combination of ability to detect trends in performance degradation (i.e., lower limit of the range of factors that might be selected), and margins to the HSS risk significance thresholds (i.e., upper limit of the range of factors that might be selected). The following provide some guidance regarding selection of an appropriate risk sensitivity factor, which may change over time.
  - Increasing the unreliability of ALL LSS SSCs by a factor of 3 to 5 provides a general indication of the potential trend in CDF and LERF, if there were a degradation in the performance of ALL LSS SSCs.
  - Such degradation is extremely unlikely for an entire group of components. The plant corrective action program would see a substantial rise in failure events and corrective actions would be taken long before the entire population experienced such degradation. In the extreme, individual components could see variations in performance on this order, but it is exceedingly unlikely that the performance of a large group of components would all shift in an unfavorable manner at the same time.
  - The risk sensitivity study should be performed by manipulating the basic event values for components that were identified in the categorization process as having low safety significance because they DO NOT support a safety-significant function. Both random and common cause PRA basic events for failure modes of the component that are relevant to the function being considered should be increased by the selected factor noted above.
  - The existing performance monitoring program must be capable of detecting a change in reliability of the LSS components by the selected factor. Standard practices used for setting performance criteria based on failures under the Maintenance Rule are applicable. This includes consideration of currently expected number of failures for the number of demands/hours of operation, and the expected number of failures for the expected future number of demands/hours of operation, for the population of SSCs that are LSS and candidate LSS. So, for example, if a factor of 3 is chosen for the risk sensitivity, the performance monitoring program must be capable of detecting an increase in unreliability for ALL LSS components by that amount. If not, a higher factor must be chosen.

5.23 {tc "5.23 Initial Sensitivity Study " \f C \l 2}Perform Initial Sensitivity Study

5.23.1 **Prepare** an initial sensitivity study for presentation to the IDP as an indication of the potential aggregate risk impacts.

5.23.2 **Perform** this sensitivity study for each individual plant system as the categorization of its functions is provided to the IDP.

5.23.3 In identifying the specific factor to be used in the risk sensitivity study, **check** that the cumulative risk increase computed with the unreliabilities of ALL previously-categorized LSS AND candidate LSS SSCs simultaneously increased by the selected factor CANNOT lead to exceeding the quantitative acceptance guidelines of Reg. Guide 1.174.

5.23.4 In cases where the categorization process identifies beyond design basis functions that will be addressed for RISC-1 (i.e., if special treatment requirements were added to address important beyond design basis functions, effectively improving the reliability of the SSC), **consider** performing a sensitivity study reducing the unreliability (i.e., increasing the reliability) of these safety-significant SSCs by a similar factor, depending upon the specific changes in special treatment, in accordance with the following factors.

- The cumulative changes in CDF and LERF computed in such sensitivity studies should be compared to the risk acceptance guidelines of Reg. Guide 1.174 as a measure of their acceptability.
- In addition, importance measures from these sensitivity studies can provide insight as to which SSCs AND which failure modes are most significant.

5.23.5 **Determine** if the recommended FV and RAW threshold values used in the screening need to be changed based on results of this sensitivity study, in accordance with the following.

- 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 **consider** using a lower F-V threshold value (e.g., > 0.0025 = HSS) for a re-evaluation of SSCs risk ranking.
- This may result in re-categorizing some of the candidate LSS SSCs as safety-significant SSCs.

5.24 {tc "5.24 Cumulative Sensitivity Study " \f C \l 2}**Perform** a cumulative sensitivity study for ALL LSS components for ALL systems that have been categorized by repeating the above process.

5.25 {tc "5.25 Cumulative Sensitivity Results to IDP " \f C \l 2}**Provide** the results of individual system AND cumulative sensitivity studies to the IDP. This should provide the IDP with both the overall assessment of the potential risk implications AND the relative contribution of each system.

5.26 {tc "5.26 Re-Evaluation of Sensitivity Studies " \f C \l 2}**Re-evaluate** sensitivities after IDP consideration

5.26.1 IF the IDP has changed SSC categorizations, THEN **check** the sensitivity studies AND **revise**, if necessary, to assure that the conclusions regarding the potential aggregate impact have NOT changed significantly.

5.26.2 IF the categorization of SSCs is done at different times, THEN **consider**, in the sensitivity study, the potential cumulative impact of ALL SSCs categorized, NOT individual systems or components.

NOTES

- A planned and phased implementation of SSC categorization over several years could result in later SSC categorization activities impacting earlier SSC categorization schemes. Thus, a review of the impact of the current categorization activity on previous categorizations should be performed.
-
- A determination needs to be made whether the integrated sensitivity study or the defense in depth implication considerations in previous categorizations have been changed as a result of these later categorization activities.
    - IF such changes are found, THEN they should be presented to the IDP for consideration in their deliberations on the categorization of the latest system. This review of previous categorization may be focused to those SSCs affected by the categorization of additional functions, and does not obviate or replace the need for periodic reviews.

5.27 {tc "5.27 PRA Periodic Reviews " \f C \l 2}**Perform** PRA Reviews to ensure continued validity of categorization results AND to review SSC performance, in accordance with the following.

5.27.1 Following each periodic update of the PRA (at least once per every other refueling outage for Unit 1), **perform** a review of ALL SSCs that have been categorized, to evaluate changes to plant design, operational practices, and industry and plant operational experience for impact on existing categorizations. The PRA update should address significant changes in operating experience for categorized SSCs, where appropriate.

5.27.2 **Consider** performing additional reviews, in addition to the periodic reviews, if a PRA model or other risk information is upgraded (as defined in Ref. 7.11), in order to determine if previously-performed categorization results are affected by the model changes.

5.27.3 **Refer** to the following to gain an understanding of the potential impacts of PRA model changes on SSC categorizations.

- In most cases, the categorization would be expected to be unaffected by changes in the plant-specific risk information. However, in some instances, an updated PRA model could result in new RAW and F-V importance measures that are significantly different from those in the original categorization. Although this would suggest a potential change in the categorization, it is important to recognize that RAW and F-V are relative (to total CDF or LERF) importance measures, such that a decrease in CDF or LERF might result in an increase in relative importance of an SSC, and vice-versa. In these cases, the assessment of whether a change in categorization is appropriate should be based on the absolute value of the importance measures.
- The absolute importance is the product of the base CDF/LERF and the importance measure ([RAW-1] or Fussell-Vesely). This is done in order to not inadvertently assess an SSC as safety significant when it's relative importance (FV and RAW) has gone up only due to a decrease in overall CDF/LERF. Consider the following *examples*:

- A PRA model change has resulted in an increase in at-power CDF. A component previously categorized as HSS now no longer meets the F-V and RAW criteria for HSS according to the new CDF or LERF values. This would suggest potential re-categorization consideration (to LSS) by the IDP. However, this would only be appropriate if the updated absolute importance measures were also below the HSS threshold. If the updated absolute importance measures indicate HSS, then the component remains HSS.
- A PRA model change has resulted in a decrease in at-power CDF. A component previously categorized as LSS now meets the F-V and RAW criteria for HSS according to the new CDF or LERF values. This would suggest potential re-categorization to HSS after consideration by the IDP. However, this would only be appropriate if the updated absolute importance measures were also above the HSS threshold. If the absolute importance measures are NOT above the threshold, this is an indication that the relative importance has increased only as a result of the reduction in CDF or LERF (i.e., an indication of an overall safety improvement), so a change in categorization would NOT be indicated.

5.27.4 IF a change to the categorization of an SSC is suggested by a change in the PRA model as determined from the absolute importance measures, THEN **present** such changes to the IDP for concurrence.

5.28 {tc "5.28 Availability of New PRA models for Risk Contributors " \f C \l 2}Availability of New PRA models for Risk Contributor(s)

NOTE

When new PRA models are developed for additional risk contributors (e.g., seismic, other external events, shutdown, etc.) AND approved for use in 50.69 categorization, it is NOT necessary to re-categorize systems that have already been categorized using appropriate qualitative analysis (e.g., SMA for seismic risk, Shutdown DID for shutdown risk, etc.) UNLESS the results of the new PRA models indicate that the risk importances of previously categorized component modeled in the new PRA exceed the criteria for candidate HSS as specified later in this section. Use the following guidance to determine if a system that was already categorized using a qualitative analysis should be re-categorized using newly-developed models for other risk contributors.

5.28.1 **Review** the set of CDF and LERF basic event importances from the new risk contributor PRA to determine if there are any previously-categorized components for which the new basic event importances exceed the criteria for HSS.

5.28.2 IF the new risk contributor PRA basic event importances for any previously categorized components exceed the criteria for HSS, THEN **determine** the integrated risk importance for those components following the process defined in Sections 5.8 through 5.26.

NOTE

Re-categorization is NOT required for systems with components whose new risk contributor PRA basic event importances do not meet the criteria for HSS, OR whose integrated risk importance evaluation does not meet the criteria for HSS. However, it may be beneficial to re-categorize these particular components if the risk contribution is lowered.

5.28.3 IF following the integrated risk importance evaluation, the component(s) still meet the criteria for candidate HSS, THEN **re-categorize** the systems associated with these components.

6.0 {tc "6.0 Records" \f C \l 1}Records

Results generated by this instruction are considered QA records. **Store** these records per NMP-ES-065-003

7.0 {tc "7.0 References" \f C \l 1}References

7.1 10 CFR 50.69, "Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors"

7.2 NEI 00-04, "10 CFR 50.69 SSC Categorization Guide, Revision 0"

7.3 NRC Regulatory Guide 1.201, "Guidelines For Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Rev 1 (for Trial Use), May 2006

7.4 NMP-ES-065, 10 CFR 50.69 Program

7.5 NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization

7.6 NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components

7.7 NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

7.8 NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

7.9 EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments"

7.10 NRC Regulatory Guide 1.200, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Rev 2, March 2009

7.11 RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", Addenda to ASME/ANS RA-S–2008, ASME/ANS, 2009.

8.0 {tc "8.0 Commitments" \f C \l 1}Commitments

None

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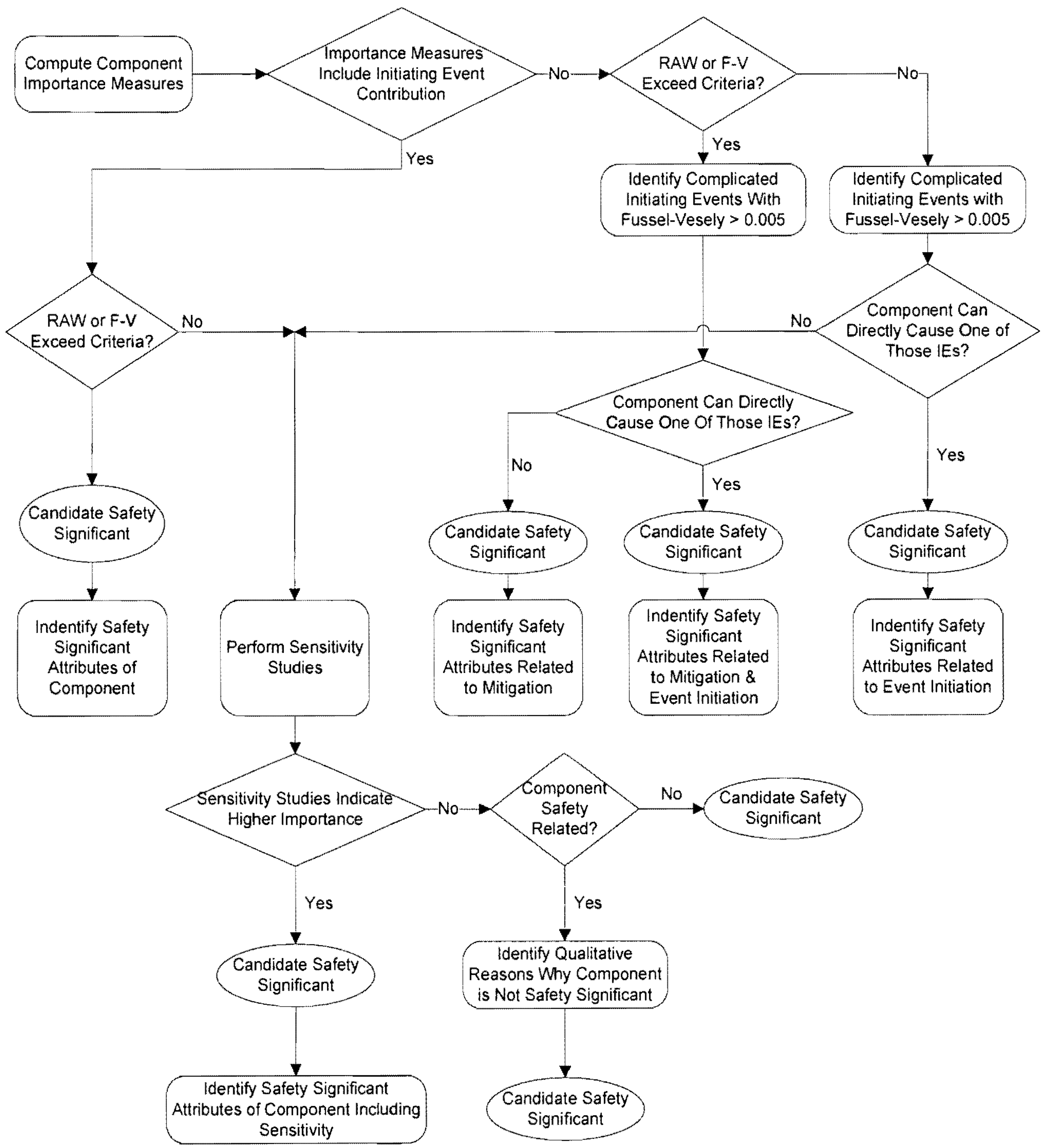
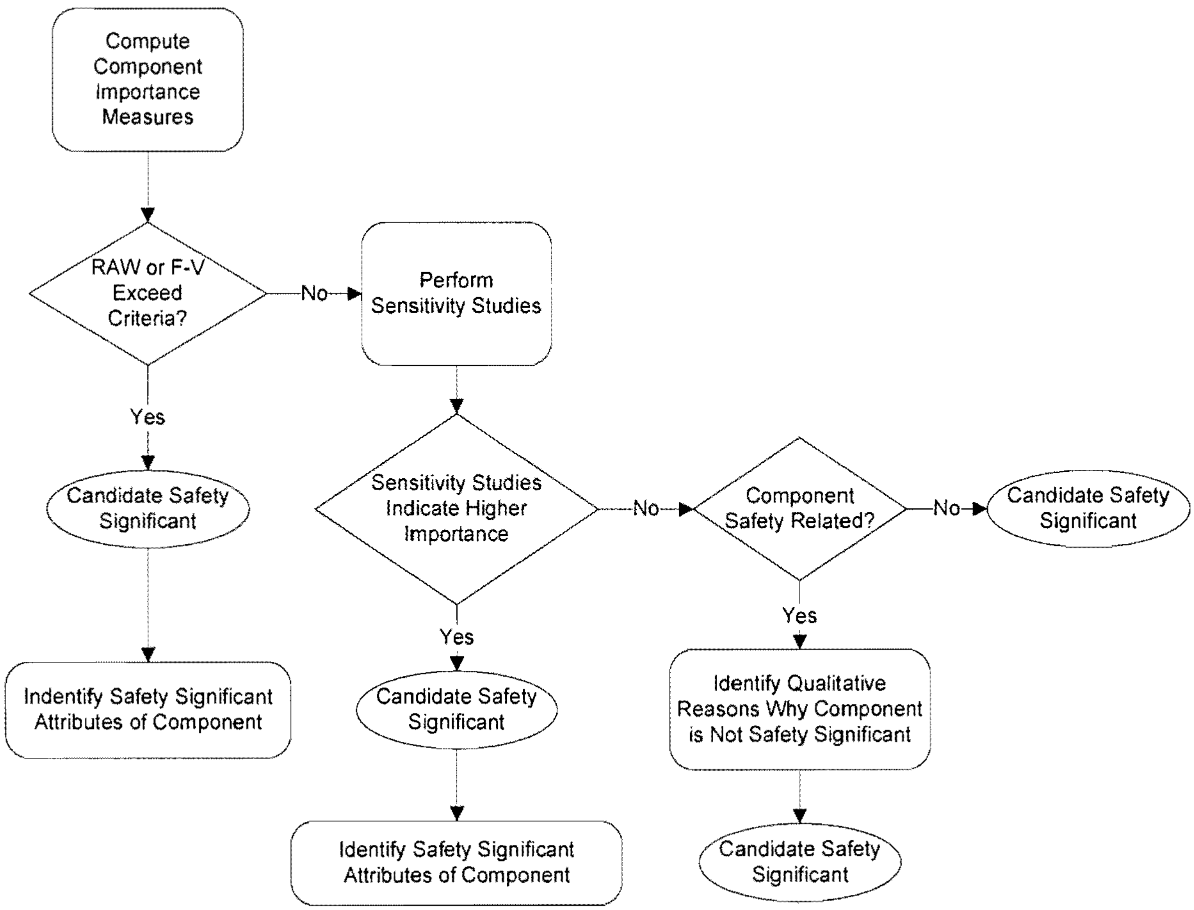
Attachment 1

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



Attachment 2

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



Instruction Owner: Amir Afzali / Risk-Informed Engineering Department Director / Corporate
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Approved By: Owen Scott for Amir Afzali / 11/23/11
(Peer Team Champion/Procedure Owner's Signature / Date)

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PRB Required

This NMP is under the oversight of the Risk-Informed Engineering Department

Writer(s): Vish Patel

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Instruction Version Description

Version Number	Version Description
1.0	Initial Issue

Table of Contents

Page

1.0 Purpose{tc "1.0 Purpose" \f C \l 1}

1.1 This instruction provides guidance to support the categorization of structures, systems, and components (SSCs) that perform a passive function, in accordance with 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.*

1.2 This instruction is part of an integrated categorization process which includes the following additional procedures/instructions:

- NMP-ES-065, 10 CFR 50.69 Program
- NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
- NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Structures, Systems, and Components
- NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

2.0 Applicability{tc "2.0 Applicability" \f 533333347 \l 1}

2.1 This instruction is applicable only to those plant systems that have been selected for categorization and which contain passive components or active components that perform a passive function.

2.2 This instruction is applicable to activities involving the 10 CFR 50.69 Passive Component Categorization performed by Southern Nuclear Operating Company (SNC) personnel or supplemental personnel.

3.0 Definitions{tc "3.0 Definitions" \f C \l 1}

All definitions are contained in NMP-ES-065. This instruction SHALL be used with NMP-ES-065.

4.0 Responsibilities{tc "4.0 Responsibilities" \f C \l 1}

Responsibilities for the 10 CFR 50.69 Process, including the activities described in this instruction, are found in NMP-ES-065.

5.0 Procedure{tc "5.0 Procedure" \f C \l 1}

NOTES

- This procedure has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Activities described in this procedure may be performed prior to NRC approval of the license amendment.
- The alternative treatment requirements specified in 10 CFR 50.69 (d) SHALL NOT be implemented UNLESS step 5.1, step 5.2, and, if applicable, step 5.3 are verified to be complete.

5.1 {tc "5.1 through 5.3 - License Amendment Review" \f C \l 2}After the license amendment is approved by the NRC, **perform** a documented evaluation to ensure that the process described in this procedure meets the requirements of, and is consistent with, the NRC-approved license amendment.

5.2 **Track** the performance of this evaluation via a Condition Report action. This evaluation SHALL be approved by the Director, Risk-Informed Engineering and by the Director, Licensing. After approval of the evaluation, **revise** this procedure to remove the above Note, this step and steps 5.1 and 5.3.

5.3 IF the above evaluation concludes that the process described in this procedure DOES NOT meet the requirements of, or is inconsistent with, the approved license amendment, THEN **revise** this procedure accordingly and **re-perform** any evaluations or activities already performed using the revised procedural requirements.

NOTES

- The source documents for the methodology referenced in this instruction is EPRI Report TR-112657, Rev B-A.
- IF further details are needed on 1) the evaluation of operator actions and its impact on the consequence ranking; 2) the evaluation and ranking of the consequence impact groups and configurations; and/or 3) the evaluation of shutdown and external events, THEN consult EPRI Report TR-112657, Rev B-A.
- IF additional guidance needs to be provided in this instruction to incorporate EPRI Report TR-112657, Rev B-A requirements, THEN contact the Risk-Informed Engineering Department.

5.4 Refer to the following to gain an understanding of the General Requirements of the Passive Component Categorization Process:{tc "5.4 General Requirements" \f C \l 2}

5.4.1 Scope – **Apply** this process to the following:

- On a system basis, including all components and their associated supports within the selected system(s).
- Class 2, 3 and non Class systems or their associated supports (exclusive of Class CC and MC items).

5.4.2 Refer to Attachment A for an overview of this process.

5.4.3 Categorization – **Classify** components and component supports in systems subject to the evaluation contained in this instruction as either High Safety Significant (HSS) or Low Safety Significant (LSS) in accordance with Sections 5.5 and 5.6.

NOTE
Personnel may be experts in more than one discipline, but are NOT required to be experts in all disciplines.

5.4.4 Required Disciplines – **Contact** representatives from the following disciplines, as necessary, to solicit input:

- Plant Operations
- Design Engineering
- Safety or Accident Analysis

5.5 **Perform** a Consequence Evaluation in accordance with the following guidance:{tc "5.5 Consequence Evaluation " \f C \l 2}

5.5.1 General Information

- To ease the analysis and documentation burden, **consider** grouping the components into piping segments that are based on similar conditional consequence (i.e., given failure of the piping segment). To accomplish this grouping, direct and indirect effects <u>SHALL</u> be assessed for each piping segment.

- **Determine** a Consequence Category for each piping segment from the Failure Modes and Effects Analysis (FMEA) as defined in Section 5.5.2 and from the Impact Group Assessments as defined in Sections 5.5.3 through 5.5.9.

- Throughout the evaluations of Sections 5.5 and 5.6, **take** credit, where possible, for plant features and operator actions to the extent that these would <u>NOT</u> be affected by failure of the segment under consideration. To take credit for operator actions, **ensure** that the following features are available:

  - An alarm or other system feature to provide clear indication of failure,
  - Equipment activated to recover from the condition <u>MUST</u> <u>NOT</u> be affected by the failure,
  - Time duration and resources are sufficient to perform operator action,
  - Plant procedures to define operator actions, and
  - Operator training in the procedures.

- **Develop** success criteria diagrams for all relevant initiating events.

5.5.2 Failure Modes and Effects Analysis (FMEA)

Identify potential failure modes for each system or piping segment and **evaluate** their effects. This evaluation <u>SHALL</u> consider the following:

- <u>Pressure Boundary Failure Size</u> - The consequence evaluation <u>SHALL</u> be conducted for a spectrum of pressure boundary failure sizes (i.e. small to large). The failure size that results in the highest consequence ranking <u>SHALL</u> be used. In lieu of this, a small leak may be assumed provided it can be ensured that the possibility of a large pressure-boundary failure has been precluded (e.g. presence of a flow restricting orifice).

- <u>Isolability of the Break</u> - A break can be automatically isolated by a check valve, a closed isolation valve, or an isolation valve that closes on a given signal. In lieu of automatic isolation, operator action may be credited consistent with 5.5.1. When isolation is credited, the impacts of both successful isolation (e.g. loss of one train) and unsuccessful isolation (e.g. loss of two trains) shall be determined and the The highest consequence ranking <u>SHALL</u> be used.

- Indirect Effects - These include spatial effects (e.g., spray, pipe whip) and loss-of-inventory effects (e.g., draining of a tank that supports multiple functions).

- Initiating Events - Initiating events caused by the postulated piping failure are identified. The list of initiating events from the plant-specific PRA and the plant design basis may be used. For systems or piping segments that are NOT modeled, either explicitly or implicitly, in the plant-specific PRA, analysis might be required to identify applicable initiating events.

- System Impact or Recovery - The means of detecting a failure, and the Technical Specifications associated with the system and other affected systems SHALL be identified. This should include possible automatic and operator actions to prevent a loss of system function.

- System Redundancy - The existence of redundancy for accident mitigation purposes SHALL be considered.

- System Configuration - The consequence evaluation and ranking is organized into four basic consequence impact groups as discussed in Sections 5.5.4 through 5.5.7. The three corresponding system configurations for these impact groups are defined in Table 6.

5.5.3 Impact Group Assessment - General

5.5.3.1 **Classify** the results of the FMEA evaluation for each system, or portion thereof into one of three core damage Impact Groups: initiating event, system, OR combination. In addition, **evaluate** failures for their importance relative to containment performance.

5.5.3.2 **Partition** each system, or portion thereof, into one of three postulated piping failures: those that cause an initiating event, those that disable a system/train/loop without causing an initiating event, OR those that cause an initiating event and disable a system/train/loop.

5.5.3.3 **Determine** the consequence category assignment (high, medium, low, or none) for each piping segment within each impact group in accordance with the following steps.

5.5.4 Initiating Event (IE) Impact Group Assessment

5.5.4.1 **Utilize** this section when the postulated failure results in only an initiating event (e.g., loss of feedwater, reactor trip).

5.5.4.2 **Classify** the failure classified into one of four categories: high, medium, low, or none. **Assign** the initiating event category according to the following:

- The initiating event SHALL be placed in one of the Design Basis Event Categories in Table 1. All applicable design basis events previously analyzed in the Owner's updated final safety analysis report or PRA SHALL be included.
- Breaks that cause an initiating event classified as Category I (routine operation) need NOT be considered in this analysis.
- For breaks that result in Category II (Anticipated Event), Category III (Infrequent Event), or Category IV (Limiting Fault or Accident), the consequence category SHALL be assigned to the initiating event according to the conditional core damage probability (CCDP) criteria specified in Table 5.
- Differences in the consequence rank between the use of Table 1 and 5 SHALL be reviewed, justified and documented or the higher consequence rank assigned. The quantitative index for the initiating event impact group is the ratio of the core damage frequency due to the initiating event to the frequency for that initiating event in the base PRA model.

5.5.5 System Impact Group Assessment

5.5.5.1 **Utilize** this section when the postulated failure does NOT cause an initiating event, but degrades or fails a system/train/loop essential to prevention of core damage.

5.5.5.2 **Include** all safety functions supported by the segment as well as all safety functions impacted by the failure of the segment.

5.5.5.3 **Perform** the evaluation based on the following:

- Frequency of challenge that determines how often the affected function of the system is called upon. This corresponds to the frequency of events that require the system operation.
- Number of backup systems (portions of systems, trains, or portions of trains) available, which determines how many unaffected systems (portions of systems, trains, or portions of trains) are available to perform the same mitigating function as the degraded or failed system.
- Exposure time - the time the system would be unavailable before the plant is changed to a different mode in which the failed system's function is no longer required, the failure is recovered, OR other compensatory action is taken. Exposure time is a function of the detection time and completion time, as defined in the plant Technical Specification, as applicable.

5.5.5.4 **Assign** a consequence category of High, Medium, or Low, in accordance with Table 2 and the following factors:

- Frequency of challenge is grouped into design basis event categories II, III, and IV.
- Quantitative indices may be used to assign consequence categories in accordance with Table 5 in lieu of Table 2 provided the quantitative basis of Table 2 (e.g., one full train unavailability approximately 10^{-2}, exposure time) is consistent with the failure scenario being evaluated.
- Differences in the consequence rank between the use of Table 2 and 5 SHALL be reviewed, justified and documented or the higher consequence rank assigned.
- The quantitative index for the system impact group is the product of the change in conditional core damage frequency (CCDF) and the exposure time.
- For defense in depth purposes, all postulated failures leading to "zero defense" (i.e., no backup trains) SHALL be assigned a high consequence.

5.5.6 Combination Impact Group Assessment

5.5.6.1 **Utilize** this section when the postulated failure results in BOTH an initiating event and the degradation or loss of a system.

5.5.6.2 **Assign** a consequence category of High, Medium, Low, or N/A in accordance with Table 3 and the following factors:

- The consequence category is a function of two considerations:
  - o Use of the system to mitigate the induced initiating event;
  - o Number of unaffected backup systems or trains available to perform the same function.
- Quantitative indices may be used to assign consequence categories in accordance with Table 5 in lieu of Table 3 provided the quantitative basis of Table 3 (e.g., one full-train unavailability approximately 10^{-2}) is consistent with the failure scenario being evaluated.
- Differences in the consequence rank between the use of Table 3 and 5 SHALL be reviewed, justified and documented or the higher consequence rank assigned.

5.5.7 Containment Performance Impact Group Assessment

Review the previously established consequence rank (from Section 5.5.4, 5.5.5, or 5.5.6) and **adjust** the rank to reflect the pressure boundary' failure's impact on containment performance, based on the following:

- **Use** Table 4 to assign consequence categories for those piping failures that can lead to a LOCA that bypasses containment.
- **Use** the CLERP quantitative indices of Table 5 for postulated failures that DO NOT result in a LOCA that bypasses containment.

5.5.8 Shutdown Impact Assessment

5.5.8.1 **Review** the previously established consequence rank (from Section 5.5.4, 5.5.5, or 5.5.6) and **adjust** the rank to reflect the pressure boundary' failure's impact on plant operation during shutdown, based on the following steps.

5.5.8.2 IF the plant has a shutdown PRA, THEN the important initiators and systems will have already been identified for shutdown operation, and their effect on core damage and containment performance will have already been determined.

5.5.8.3 IF a shutdown PRA is NOT available, THEN the effect of pressure-boundary failures on core damage and containment performance SHALL be evaluated in accordance with the following considerations:

- The system operations, safety functions, and success criteria change in different stages of other modes of operation.
- The exposure time for the majority of the piping associated with shutdown operation is typically less than 10 percent per year. The exposure time associated with being in a more risk-significant configuration is even shorter, depending on the function or system that is being evaluated.
- The unavailability of mitigating trains could be higher due to planned maintenance activities. Shutdown guidelines need to be evaluated to assure that sufficient redundancy is protected during different modes of operation.
- Recovery time may be longer, thus allowing for multiple operator actions.

5.5.9 External Events

Review the previously established consequence rank (from Section 5.5.4, 5.5.5, or 5.5.6) and **adjust** the rank to reflect the pressure boundary' failure's impact on the mitigation of external events. **Evaluate** the effect of external events on core damage and containment performance from two perspectives, as follows.

- External events that can cause a pressure boundary failure (e.g. seismic events), and
- External events that DO NOT affect likelihood of pressure-boundary failure, but create demands that might cause pressure-boundary failure and events (e.g. fires).

5.6 Classification{tc "5.6 Classification " \f C \l 2}

NOTE
Piping segments may be grouped together within a system, if the consequence evaluation performed in Section 5.5 determines the effect of the postulated failures to be the same. The classification designations referred to in this Section are: • HSS - Piping segment considered high-safety-significant • LSS – Piping segment considered low-safety-significant

Classify each piping segment as HSS or LSS based on the following requirements:

5.6.1 IF the piping segment was determined to be a High consequence category in any table by the Consequence Evaluation in Section 5.5, THEN **classify** that segment as HSS.

NOTES
• For Section 5.5.2, it is NOT necessary to assume a large pressure boundary leak as long as the same conditions described in Section 5.4.2 for Pressure Boundary Failure Size are met. • In addition, when using this instruction, the Risk-Informed Application Engineer (or designee) may take credit for plant features and operator actions to the extent that these would NOT be affected by failure of the segment under consideration. • IF plant features and operator actions are credited, THEN they SHALL be consistent with those credited in section 5.4.1.

5.6.2 IF the piping segment was determined to be a Medium, Low, or None (no change to base case) consequence category in all tables by the consequence evaluation in Section 5.5, THEN **perform** an additional assessment by evaluating the conditions listed below. **Answer** each condition as either TRUE or FALSE.

1) Failure of the pressure retaining function of the segment will NOT directly OR indirectly (e.g., through spatial effects) fail a basic safety function.

2) Failure of the pressure retaining function of the segment will NOT prevent the plant from reaching or maintaining safe shutdown conditions; and the pressure retaining function is NOT significant to safety during mode changes or shutdown. Assume that the plant would be unable to reach or maintain safe shutdown conditions if a pressure boundary failure results in the need for actions outside of plant procedures or available backup plant mitigative features.

3) The pressure retaining function of the segment is NOT called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient.

4) The pressure retaining function of the segment is NOT called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities.

5) Failure of the pressure retaining function of the segment will NOT result in an unintentional release of radioactive material that would result in the implementation of offsite radiological protective actions.

6) This process SHALL demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth is maintained if ALL of the following criteria are met:

   - Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of an offsite release.
   - There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.
   - System redundancy, independence, and diversity are preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters.
   - Potential for common cause failures is taken into account in the risk analysis categorization.
   - Independence of fission-product barriers is NOT degraded.

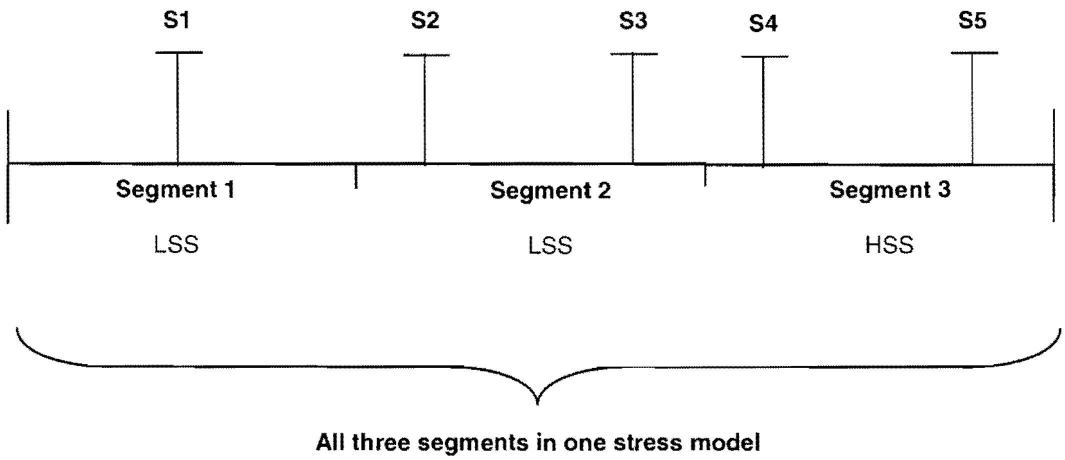
IF ANY of the above conditions are answered FALSE, THEN **classify** the associated pipe segment as HSS. Otherwise, **classify** the segment as LSS.

5.6.3 IF LSS has been assigned from Section 5.6.2, THEN **verify** that there are sufficient margins to account for uncertainty in the engineering analysis and in the supporting data in accordance with the following:

- Margin SHALL be incorporated when determining performance characteristics and parameters, e.g., piping segment, system, and plant capability or success criteria.
- The amount of margin should depend on the uncertainty associated with the performance parameters in question, the availability of alternatives to compensate for adverse performance, and the consequences of failure to meet the performance goals.
- Sufficient margins are maintained by ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or proposed revisions account for analysis and data uncertainty.

5.6.4 IF sufficient margins are maintained, THEN **retain** the LSS classification. Otherwise, **re-classify** the pipe segment as HSS.

5.6.5 **Assign** each component support, hanger, or snubber the same classification as the highest-ranked piping segment within the piping stress analytical model in which the support is included. This requirement is better explained using the following figure (as example):



The above depicts that support S1 is in pipe segment 1; supports S2 and S3 re in pipe segment 2; and supports S4 and S5 are in pipe segment 3. Segments 1 and 2 are determined to be LSS. Segment 3 is determined to be HSS. Typically, a support associated with the segment gets the same safety significance as the segment itself. For example, segment 2 is categorized as LSS; therefore, associated supports S2 and S3 will be categorized as LSS. However, in this example, the three segments were in the same stress model AND segment 3 is categorized as HSS; so all supports (S1 thru S5) SHALL be categorized as HSS regardless of LSS classification of segments 1 and 2. Note that this requirement applies to the supports only. The pipe segments maintain their original risk rank.

6.0 Records{tc "6.0 Records" \f C \l 1}

The results generated by this instruction are considered QA records. They will be stored per NMP-ES-065-003.

7.0 **References{tc "7.0 References" \f C \l 1}**

7.1 NMP-ES-065, 10 CFR 50.69 Program

7.2 NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights

7.3 NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components

7.4 NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

7.5 NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

7.6 ANO-2 SER, Safety Evaluation by the Office of Nuclear Reactor Regulation Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals, dated April 22, 2009.

7.7 NEI 00-04, Revision 0, 10 CFR 50.69 SSC Categorization Guideline, July, 2005.

7.8 10 CFR 50.69 Final Rule, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, November 22, 2004.

7.9 EPRI TR-112657, Rev B-A, Revised EPRI Risk-Informed In-service Inspection Evaluation Procedure, EPRI, Palo Alto, CA: 1999.

7.10 NUMARC 91-06, “Guidelines for Industry Actions to Address Shutdown Management” dated 1991.

7.11 NUREG-0800, Section 3.6.1 “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

7.12 NUREG-0800, Section 3.6.2 “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

8.0 **Commitments{tc "8.0 Commitments" \f C \l 1}**

None.

TABLE 1

CONSEQUENCE CATEGORIES FOR INITIATING EVENT IMPACT GROUP

Design Basis Event Category	Initiating Event Type	Representative Initiating Event Frequency Range (1/yr)	Example Initiating Events	Consequence Category (Note 1)
I	Routine Operation	>1		None
II	Anticipated Event	10^{-1}<value≤1	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	10^{-2}<value≤10^{-1}	Excessive Feedwater or Steam Removal	Low/Medium
			Loss of Off Site Power	Medium/High
IV	Limiting Fault or Accident	≤10^{-2}	Small LOCA, Steam Line Break, Feedwater Line Break, Large LOCA	Medium/ High

Note 1: Refer to 5.5.4

TABLE 2

GUIDELINES FOR ASSIGNING CONSEQUENCE CATEGORIES TO FAILURES RESULTING IN SYSTEM OR TRAIN LOSS

Affected Systems		Number of Unaffected Backup Trains							
Frequency of Challenge	**Exposure Time to Challenge**	**0.0**	**0.5**	**1.0**	**1.5**	**2.0**	**2.5**	**3.0**	**≥ 3.5**
Anticipated (DB Cat II)	All Year	HIGH	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW
	Between tests (1-3 months)	HIGH	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW
	Long CT (≤ 1 week)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Short CT (≤ 1 day)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
Infrequent (DB Cat. III)	All Year	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW
	Between tests (1-3 months)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Long CT (≤ 1 week)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short CT (≤ 1 day)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
Unexpected (DB Cat. IV)	All Year	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Between tests (1-3 months)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Long CT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short CT (≤ 1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

* - If there is no containment barrier and the consequence category is marked by an *, the consequence category should be increased (medium to high or low to medium).

TABLE 3

CONSEQUENCE CATEGORIES FOR COMBINATION IMPACT GROUP

Event	Consequence Category
Initiating Event and 1 Unaffected Train of Mitigating System Available	High
Initiating Event and 2 Unaffected Trains of Mitigating Systems Available	Medium[1] (or IE Consequence Category from Table 1)
Initiating Event and More Than 2 Unaffected Trains of Mitigating Systems Available	Low[1] (or IE Consequence Category from Table 1)
Initiating Event and No Mitigating System Affected	N/A

[1] - The higher classification of this table or Table 1 SHALL be used.

TABLE 4
CONSEQUENCE CATEGORIES FOR FAILURES RESULTING IN INCREASED POTENTIAL FOR AN UNISOLATED LOCA OUTSIDE OF CONTAINMENT

Protection Against LOCA outside Containment	*Consequence Category*
One Active[1]	HIGH
One Passive[2]	HIGH
Two Active	MEDIUM
One Active, One Passive	MEDIUM
Two Passive	LOW
More than Two	NONE

[1] - An example of Active Protection is a valve that needs to close on demand.
[2] - An example of Passive Protection is a valve that needs to remain closed.

TABLE 5
QUANTITATIVE INDICES FOR CONSEQUENCE CATEGORIES

CCDP, no units	CLERP, no units	Consequence Category
$>10^{-4}$	$>10^{-5}$	High
$10^{-6} <$ value $\leq 10^{-4}$	$10^{-7} <$ value $\leq 10^{-5}$	Medium
$\leq 10^{-6}$	$\leq 10^{-7}$	Low
No change to base case	No change to base case	None

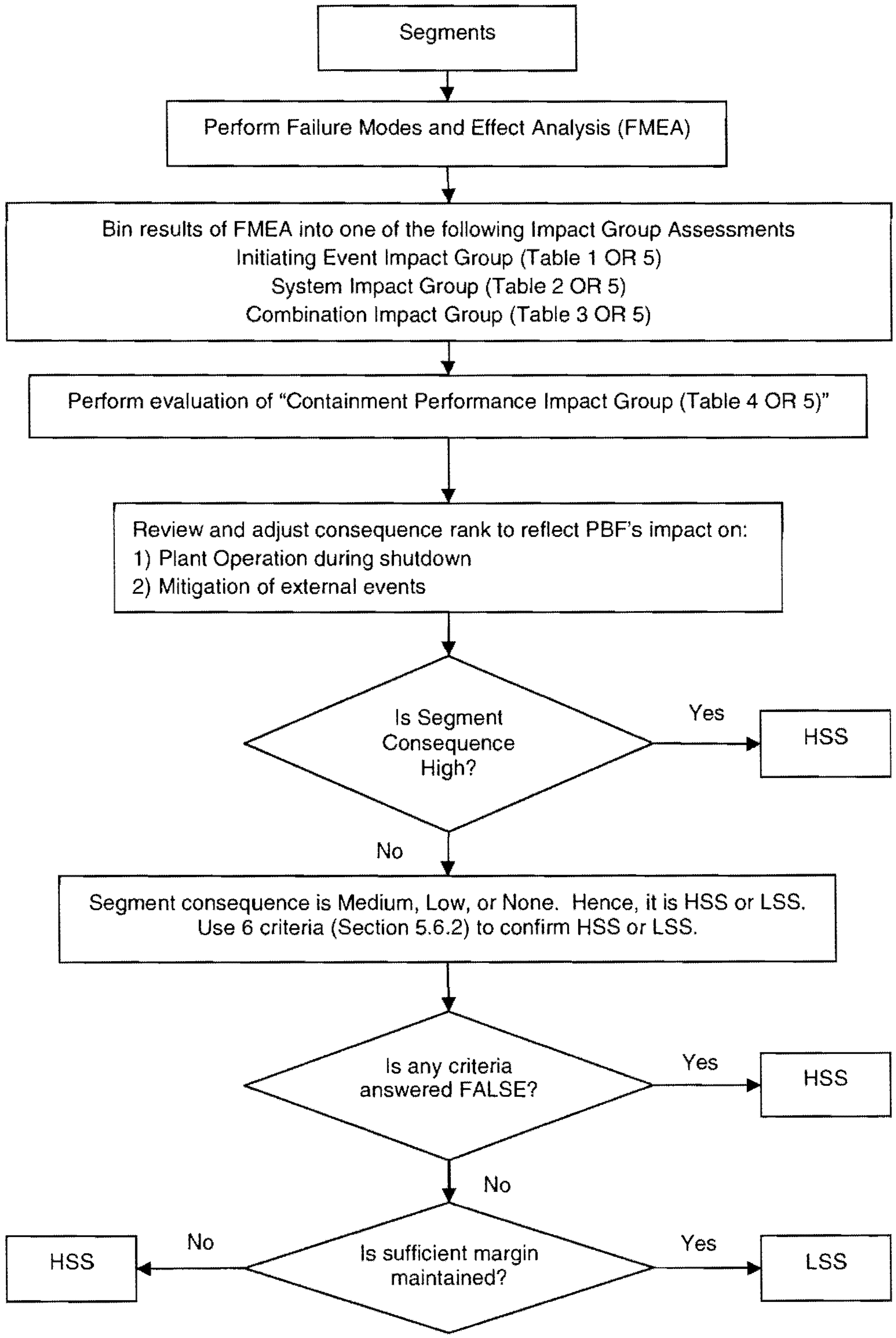
Table 6
Definition of Consequence Impact Groups and Configurations

CONSEQUENCES		
Impact Group	**Configuration**	**Description**
Initiating Event	Operating	A PBF* occurs in an operating (pressurized) system resulting in an initiating event
Loss of Mitigating Ability	Standby	A PBF occurs in a standby system and does <u>NOT</u> result in an initiating event, but degrades the mitigating capabilities of a system or train. After failure is discovered, the plant enters the applicable Allowed Outage Time defined in the Technical Specification, as applicable
	Demand	A PBF occurs when system/train operation is required by an independent demand
Combination	Operating	A PBF causes an initiating event with an additional loss of mitigating ability (in addition to the expected mitigating degradation due to the initiator)
Containment	Any	A PBF, in addition to the above impacts, also affects containment performance

PBF – pressure-boundary failure

Attachment A

Passive Component Categorization Process



Instruction Owner: Amir Afzali / Risk-Informed Engineering Department Director / Corporate
(Print: Name / Title / Site)

Approved By: Owen Scott for Amir Afzali / 11/23/11
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:	11/23/2011	N/A	N/A	11/23/2011	N/A
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PRB Required

This NMP is under the oversight of the Risk-Informed Engineering Department

Writer(s): Vish Patel

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Revision Description

Version Number	Revision Description
1.0	Initial issue

Table of Contents

1.0 { TC "1.0 Purpose" \f C \l "1" }Purpose

1.1 This instruction provides guidance to support the categorization of structures, systems, and components (SSCs) in accordance with 10 CFR 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors*.

1.2 This instruction is part of an integrated categorization process which includes the following additional procedures/instructions:

- NMP-ES-065, 10 CFR 50.69 Program
- NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
- NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization
- NMP-ES-066-002, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

1.3 The process described in this instruction and the above-listed procedures/instructions satisfies the requirements of 10 CFR 50.69 (c), *SSC Categorization Process*, and (e), *Feedback and Process Adjustment.* The scope of this instruction does NOT include alternative treatment requirements specified in 10 CFR 50.69 (d).

1.4 The process described in this instruction is consistent with Nuclear Energy Institute industry guidance document, NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline,* Rev. 0.

2.0 { TC "2.0 Applicability" \f C \l "1" }Applicability

This instruction is applicable only to those plant systems that have been selected for categorization. Since 10 CFR 50.69 is a voluntary rule, each Site may decide which plant systems to categorize or not categorize. However, once a system is selected for categorization, ALL the components in that system MUST be included in the categorization process.

3.0 { TC "3.0 Definitions" \f C \l "1" }Definitions

All definitions are contained in NMP-ES-065. This instruction SHALL be used with NMP-ES-065.

4.0 { TC "4.0 Responsibilities" \f C \l "1" }Responsibilities

4.1 IDP (Integrated Decision Making Panel)

- Evaluates PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization for system functions and components.
- Reviews results from performance monitoring and periodic reassessments to ensure that the basis for the categorization of SSCs remains valid and that any implemented alternative treatments have NOT significantly degraded the performance of the associated components.
- Evaluates recommended changes to categorization results due to PRA model updates, changes to the plant, changes to operational practices, as well as other applicable changes.

4.2 Cognizant Risk-Informed Application Engineer

- Establishes, in concert with Site Management, the criteria for and the selection of plant systems to be categorized.
- Provides the PRA base case risk and results of sensitivity studies for SSCs in the system under review, as further detailed in Reference 7.4.
- Provides the results of other hazards analyses for those hazards that are NOT modeled in the PRA, as further detailed in Reference 7.4.
- Provides additional PRA Model insights which may influence the SSC categorization outcome.
- Provides PRA risk insights in support of the passive risk categorization of SSCs, as further detailed in Reference 7.5.
- Communicates PRA risk changes, resulting from model updates or other factors that could impact existing SSC categorizations.

4.3 Site Management

- Provides input in establishing the criteria for and the selection of plant systems to be categorized
- Provides the needed resources to support the categorization effort, including:
    - Applicable IDP members
    - System Engineer
    - Operations Representative
    - Supporting material such as drawings, design criteria, procedures, etc.

4.4 Cognizant Licensing Engineer

- Assesses the system under review for regulatory or commitment insights which may influence the SSC categorization outcome.

4.5 Cognizant System Engineer

- Develops system functions
- Maps each component in the system to the system function(s) supported
- Assesses system health or equipment performance insights which may influence the SSC categorization outcome.
- Provides insights on relevant industry-related performance issues which may influence the SSC categorization outcome.
- Participates in the categorization of SSCs in the assigned system.

4.6 Operations Representative

- Provides draft qualitative responses to the essential questions used to assess the risk of system functions.
- Participates in the categorization of SSCs

4.7 Safety Analysis Representative

- Ensures that defense-in-depth for core damage, large early release, and long term containment integrity is preserved in accordance with the guidelines provided in Attachment 1.
- Participates in the categorization of SSCs.
- Ensures that sufficient safety margins are maintained for RISC-3 components.

NOTE

Because the only requirements that are relaxed for LSS SSCs are those related to treatment, existing safety margins for SSCs arising from the design technical and functional requirements would remain. Consequently, no specific assessment of safety margin is required (Ref. 7.2).

4.8 Design Engineering Representative

- Ensures that system functions are accurate and complete, including design basis functions and beyond design basis functions credited in the PRA.
- Ensures that component critical attributes are appropriately identified in relation to their role in the safety significant functions of the component.
- Participates in the categorization of SSCs

5.0 { TC "5.0 Procedure" \f C \l "1" }Procedure

NOTES

- This procedure has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Activities described in this procedure may be performed prior to NRC approval of the license amendment.
- The alternative treatment requirements specified in 10 CFR 50.69 (d) SHALL NOT be implemented UNLESS step 5.1, step 5.2, and, if applicable, step 5.3 are verified to be complete.

5.1 {tc "5.1 through 5.3 - License Amendment Review" \f C \l 2}After the license amendment is approved by the NRC, **perform** a documented evaluation to ensure that the process described in this procedure meets the requirements of, and is consistent with, the NRC-approved license amendment.

5.2 **Track** the performance of this evaluation via a Condition Report action. This evaluation SHALL be approved by the Director, Risk-Informed Engineering and by the Director, Licensing. After approval of the evaluation, **revise** this procedure to remove the above Note, this step and steps 5.1 and 5.3.

5.3 IF the above evaluation concludes that the process described in this procedure DOES NOT meet the requirements of, or is inconsistent with, the approved license amendment, THEN **revise** this procedure accordingly and **re-perform** any evaluations or activities already performed using the revised procedural requirements.

NOTE

Appropriate steps in the following process are to be documented, including the basis. As applicable, this documentation should be entered into a database and coded where practical in order to facilitate data manipulation and retrieval tasks.

5.4 { TC "5.4 Categorization Process Elements" \f C \l "2" }**Refer** to the following to gain an understanding of the essential elements of the Categorization Process:

5.4.1 Risk Categories

SSCs SHALL be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 using the categorization process outlined in this instruction that determines the functions that an SSC performs or supports and if any of those functions are HSS.

5.4.2 PRA Capability

NOTE

Additional details on PRA capability requirements are provided in Reference 7.4.

The risk-informed categorization of SSCs in nuclear power plant applications requires the use of an appropriately detailed PRA of sound technical quality. In evaluating the PRA, the following factors are to be considered:

- At a minimum, the PRA MUST model severe accident scenarios resulting from internal initiating events occurring at full power operation.
- PRA limitations may include hazards that are NOT modeled (e.g., external initiating events), plant shutdown risks, and SSCs that are NOT modeled.
- These limitations can be addressed through supplementary analyses. Typically, these involve bounding analyses or qualitative methods such as screening assessments and/or IDP evaluations.

5.4.3 Qualitative Insights

NOTES

- Due to PRA assumptions and limitations, such as those mentioned above, qualitative insights are used to supplement the PRA risk results.
- Qualitative insights are typically needed to categorize components within a particular plant system, primarily because many components in a particular system are not modeled by the PRA.
- These insights can provide an alternate and valuable perspective that can be blended with the PRA results to reach an overall risk assessment.

Qualitative insights include, but are not necessarily limited, to the following:

- Supplementary analyses that are used to compensate for PRA limitations in quantifying the risk during plant shutdown and for hazards that may not be modeled such as fire risks, seismic risks, and other external risks (e.g., tornadoes, external floods, etc.)
- Qualitative risk assessment that considers, like the PRA, the impact and likelihood of failure of the SSC under consideration.
- Plant design bases
- Maintenance of defense-in-depth
- Maintenance of sufficient safety margins
- Plant and industry operating experience
- Operational and maintenance processes

5.4.4 Passive (Pressure Retention) Risk of Components

> **NOTE**
> Additional details on passive risk are provided in Reference 7.5

Components (including associated supports) having only a pressure retaining function (also referred to as passive components), OR active components having a passive function are required to undergo a separate passive risk assessment process, in accordance with the following:

- This process is based on the EPRI risk-informed in-service inspection (RI-ISI) evaluation, supplemented by additional qualitative considerations.
- Each piping component (including valves and supports) is categorized as HSS or LSS based on the consequence evaluations of an assumed pressure boundary failure.
- The consequence evaluations use both PRA and qualitative insights.
- Although all ASME component classes can be categorized using this process, it should be noted that alternative treatments to ASME Section XI for repair/replacement activities can only be applied to ASME Class 2 and 3 pressure retaining items or their associated supports.

5.4.6 Overall Categorization

SSCs that are considered HSS based on PRA results, qualitative results, OR evaluation of passive risk (if applicable), SHALL be categorized as RISC-1 or RISC-2. Otherwise, they can be categorized as RISC-3 or RISC-4.

5.4.7 Integrated Decision Making Panel

> **NOTE**
> Additional details on the IDP are provided in Reference 7.7.

SSC categorization SHALL be performed by an IDP, staffed with expert, plant-knowledgeable members. For the purpose of the categorization process, the expertise of the IDP members SHALL include, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering. The IDP evaluates PRA risk results along with qualitative insights and defense-in-depth considerations to arrive at consensus-based categorization decisions.

5.4.8 Training

Specific training and qualifications requirements for IDP members and designated alternates is detailed in Reference 7.7. Familiarity training on the categorization process should also be provided to other individuals who may participate in the IDP meetings, such as the cognizant system engineer for the system under discussion.

5.4.9 Scope of SSC categorization

The categorization process is a voluntary process that may be applied to selected plant systems or structures. However, ONCE a system selection is made, THEN ALL the components within the system or structure are to be categorized, NOT just specific components within a system or structure. The categorization scope for a particular system or structure includes ALL components possessing a unique component identification number in the Plant Data Management System (PDMS) AND identified in PDMS as belonging to that system.

5.4.10 Periodic Reviews and Performance Feedback

For those SSCs that have been categorized, periodic reviews SHALL be conducted to ensure continued validity of categorization results and to review SSC performance. Changes to plant design, operational practices, and industry and plant operational experience should be evaluated for impact on existing categorizations.

5.5 { TC "5.5 Selection of Plant Systems to be Categorized" \f C \l "2" }**Select** Plant Systems to be categorized in accordance with the following:

5.5.1 **Establish** selection criteria to help in identifying the list and sequence of systems to be categorized. Factors to consider include but are not limited to expected benefits, PRA capability to support, plant priorities, and system health and reliability.

5.5.2 **Postpone** the categorization of support systems (e.g., cooling systems or electrical distribution systems) until the majority, if not all, of the supported systems are first categorized. This will allow the risk of individual SSC loads to be determined first which can then be used to assess the risk of the supporting SSCs.

5.6 { TC "5.6 Collection of System Functional Information" \f C \l "2" }**Collect** and **Assemble** System Functional Information

5.6.1 **Identify** the system to be categorized

5.6.2 **Develop** a list of functions performed by the system using the following guidance:

- **Identify** ALL functions, NOT just those that are perceived to be safety significant. This will ensure a complete understanding of the role of the system and its interfaces with other systems.
- **Review** available sources of information for the development of system functions including, but not necessarily limited to Maintenance Rule functions, design basis documents, system descriptions, Piping and Instrumentation Diagrams (P&IDs), and the Final Safety Analysis Report (FSAR).

5.6.3 **Assign** a unique identification number to each function. The system designator should be embedded in the function number.

5.6.4 **Identify** the components within the system using the following guidance:

- **Include** ALL the components that are uniquely identified on the P&ID(s) or the single line diagrams associated with the system and designated as being part of the system.
- **Utilize** the PDMS to identify ALL active (i.e., not spared, deleted, or retired) components that are associated with the system of interest and to retrieve applicable information about each component such as:
  - Service description
  - Safety classification
  - Equipment type
- **Ensure** that piping segments and supports/snubbers are also included in the list of components that are uniquely identified

5.6.5 For each component, **identify** the system function(s) that the component supports, as follows:

- The same sources of information utilized for development of system functions can be used for this task, supplemented, as applicable, by PRA information about the component.
- In some cases, an individual component may support a function in another system. For example, a heat exchanger may belong to the cooling system but obviously supports the cooled system as well.
- Each component SHALL be associated with at least one system function. There may be cases where a new system function must be developed and added to the list of functions to account for a particular component.

5.7 { TC "5.7 Collection of System Operational Information" \f C \l "2" }**Collect** and **Evaluate** System Operational Information

5.7.1 **Collect** plant and industry operating experience relevant to the system or its components using the following guidance:

- **Focus** on equipment failures or significant degradations and review for importance, commonality, and repeat occurrences.
- **Identify** any SSCs that exhibit poor performance.
- **Summarize** the evaluation for presentation to the IDP and identify any potential categorization or treatment impacts.

5.7.2 **Identify** the current (18 months) and historical (past five years) Maintenance Rule (MR) information for the system, including MR status, unreliability and unavailability data, if applicable, and any exceedances of performance criteria.

5.7.3 **Review** licensing commitments for the system or its components and identify any commitments that could impact categorization or treatment.

5.8 { TC "5.8 Assessing Risk Hazards" \f C \l "2" }**Refer** to the following to gain an understanding of the process for assessing risk hazards using the PRA or other analyses:

> **NOTE**
>
> Specific guidance for the use of PRA or other analyses to assess risk hazards is provided in Reference 7.4.

5.8.1 The categorization process requires the assessment of a full scope of hazards consisting of:

- Internal Events Risks, including internal flooding
- Fire Risks
- Seismic Risks
- Other External Risks (e.g., tornadoes, external floods, etc)
- Shutdown Risks

> **NOTE**
>
> Components that are NOT PRA-modeled (either explicitly or implicitly) are presumed to be neither LSS or HSS but are passed through for consideration by the other portions of the process (i.e., passive risk, qualitative risk, and non-modeled hazards evaluations, as applicable).

5.8.2 The process for assessing these risk hazards is detailed in Reference 7.4 and is consistent with Reference 7.2. This process generates the following risk assessment results to be used as input into the overall categorization of SSCs, as detailed in this procedure:

- For any of the above hazards that are NOT modeled in the PRA, the results of the hazards evaluations (bounding, qualitative, or screening) that indicate which components are considered HSS.
- For components that are modeled by one or more PRAs, the individual model and integrated importance assessments (i.e., PRA risk, Fire Risk, if modeled) of LSS or HSS for each such component.
- For modeled components that are identified as having a PRA risk of LSS, the results of the required sensitivity studies.
- Modeled components that are identified as having a PRA risk of LSS and are within 10% of any of the thresholds for HSS (referred to as buffer zone components.

5.9 { TC "5.9 Passive Risk Assessment" \f C \l "2" }**Refer** to the following to gain an understanding of the Passive Risk Assessment Process:

NOTE
Specific guidance for Passive Risk Assessment is provided in Reference 7.5.

The passive risk (also known as pressure retention risk) for applicable components (i.e., pressure retaining components) in the system being categorized SHALL be determined through the process detailed in Reference 7.5. The following is a summary of this process as it relates to the overall categorization process:

- The passive risk of ASME Class 1 components SHALL be HSS.
- The Passive Risk Assessment Process generates, as an input to the overall categorization, a passive risk of either HSS or LSS for applicable components.
- A component support, hanger, or snubber SHALL have the same risk as the passive risk of the highest ranked piping segment within the piping analytical model in which the support is included.
- Other non-piping components that support a pressure retention function (e.g., valves) SHALL be assigned the same passive risk as the highest ranked piping on either side of the component.

5.10 { TC "5.10 Qualitative Risk Assessment " \f C \l "2" }**Perform** a Qualitative Risk Assessment of System Functions and Components

5.10.1 **Answer** each of the following questions for each function. IF ANY of the answers are "YES", THEN **assign** the function a risk of HSS. IF ALL of the answers are "NO", THEN **assign** the function a risk of LSS.

- Does failure of the function directly cause an initiating event?
- Does failure of the function cause a loss of reactor coolant pressure boundary integrity resulting in leakage beyond normal makeup capability?
- Does failure of the function result in the failure of a basic safety function?
- Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient? This also applies to instrumentation and other equipment needed to allow the required actions to be performed.
- Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities? This also applies to instrumentation and other equipment needed to allow the required actions to be performed.
- Does failure of the function prevent the plant from reaching or maintaining safe shutdown conditions and/or is the function significant to safety during mode changes or shutdown? Assume that the plant would be unable to reach or maintain

safe shutdown conditions if the function failure results in the need for actions outside of plant procedures or available backup functions/SSCs.

- Does failure of the function that acts as a barrier to fission product release during plant operation or during severe accidents result in the implementation of off-site radiological protective actions?

5.10.2 **Evaluate** each component in the system in accordance with the following:

> **NOTE**
> This section excludes component passive risk, which is discussed in Section 5.9 and in Reference 7.5.

5.10.2.1 **Assign** the component an initial qualitative risk based on the highest risk of any function supported by that component. For example, if the component supports two functions, one being HSS and the other LSS, the component would be assigned an initial qualitative risk of HSS.

5.10.2.2 **Evaluate** the possibility of changing the initial risk of HSS to LSS <u>IF</u> the failure of the component would <u>NOT</u> preclude the fulfillment of the HSS function. Specific considerations include, but are not limited to:

- There is no credible failure mode for the component that would prevent an HSS function from being fulfilled (e.g., a locked open or locked closed valve, a manually controlled valve, etc.),
- A failure of the component would <u>NOT</u> prevent an HSS function from being fulfilled (e.g., a vent or drain line that is not a significant flow diversion path, components downstream of the first isolation valve from the active pathway of the function, etc.), and
- Instrumentation that would <u>NOT</u> prevent an HSS function from being fulfilled (e.g., radiation monitors that do not have a direct diagnosis function, etc.).

5.10.2.3 **Exercise** caution and conservative judgment before such risk reduction allowances can be taken. **Ensure** that appropriate justification is documented.

5.11 { TC "5.11 Overall Risk Assessment " \f C \l "2" }**Determine** the Overall Risk Assessment of Components

5.11.1 <u>IF</u> <u>ANY</u> of the following assessments indicate that a component should be HSS, <u>THEN</u> **assign** that component a <u>preliminary</u> risk of HSS. Otherwise, **assign** the component a <u>preliminary</u> risk of LSS.

- Evaluation results for modeled hazards (from section 5.8 and Reference 7.4)
- Evaluation results for non-modeled hazards (from section 5.8 and Reference 7.4)
- Passive risk (from section 5.9 and Reference 7.5)
- Qualitative risk (from section 5.10.2)

5.12 { TC "5.12 Defense in Depth Assessment " \f C \l "2" }**Perform** a Defense in Depth Assessment

5.12.1 For components whose overall risk is LSS from Section 5.11, **assess** the role of the components in preserving defense-in-depth related to core damage, large early release and long term containment integrity, in accordance with the methodology provided in Attachment 1.

5.12.2 IF the defense-in-depth assessments for EITHER core damage or containment integrity CANNOT confirm the low safety significance of a particular component, THEN **re-categorize** the component as preliminarily HSS. Otherwise, **maintain** its risk of LSS.

5.13 { TC "5.13 Compiling Risk Evaluation Data " \f C \l "2" }**Compile** the following Risk Evaluation Data (for the selected system and its associated components) for presentation to the IDP:

- Licensing commitment review
- Qualitative risk results for system functions
- Operating experience review
- Assessment of system health and equipment performance
- PRA individual model and integrated risk assessments for modeled components
- Evaluation results for non-modeled hazards
- Results of PRA sensitivity studies for any of the PRAs used
- PRA LSS components that are in the buffer zone
- Passive risk for applicable components
- Qualitative risk results for system components
- Defense-in-depth assessments

5.14 { TC "5.14 IDP Evaluation " \f C \l "2" }IDP Evaluation

> **NOTE**
>
> The IDP SHALL evaluate the risk results and other system information and develop a consensus on the risk categorization of the system functions and components using the following guidance.

5.14.1 **Refer** to the following to gain an understanding of the general requirements:

- The intent of the IDP review is to ensure that SSCs have been appropriately categorized with a documented supporting basis.
- The IDP may request personnel with additional expertise or information be present at the meeting to facilitate completion of the categorization.
- The IDP does NOT need to verify the complete mapping of components to the function being evaluated. This is because IF the system function is found to be HSS, THEN ALL components supporting the function are initially considered to be HSS.
- IF a detailed categorization is performed after the initial categorization, THEN the results are separately reviewed by the IDP. This same criteria as the initial categorization is applied.
- For HSS SSCs, IF the categorization is appropriate, THEN the IDP CANNOT move the SSC to an LSS category.

5.14.2 **Ensure** that system functions satisfy the following requirements:

- System Functions should completely describe the system.
- System Functions should be categorized in a sound, consistent, and well documented manner.
- The answer to each essential question should be supported by an appropriate basis.
- The answers are reasonable and consistent, both within the selected system and, as other systems are categorized, across systems.

5.14.3 **Ensure** the following aspects of the categorization process are understood:

- PRA results for modeled components, including any assumptions or limitations. Where there are separate PRAs (e.g., Internal and Fire), the results, as presented to the IDP, should have already been integrated as previously described and as detailed in Reference 7.4.
- Evaluation results for non-modeled hazards (e.g., seismic risk), with specific attention to scope, assumptions, and degree of conservatism to the extent that the analyses point to a higher risk than the PRA base case results.
- Sensitivity results including the base and integral risk for each hazard.
- Passive risk results including assumptions and use of bounding assessments.

5.14.4 **Evaluate** qualitative risk results for components with particular attention to:

- Cases where an LSS component supports an HSS function
- Components that provide support for another system
- Risk of inadvertent actuation
- Consistency within a group of related components (e.g., air operated valve, associated solenoid valve, associated actuating sensor).

5.14.5 **Confirm** defense in depth and safety margins considerations for safety related LSS components through the following factors:

- The results of the sensitivity study that increases the failure rate of PRA-modeled components show that the increase in CDF and LERF to be sufficiently small.
- The contribution of an SSC to prevention of initiating events and to mitigation of accidents is sufficiently small.
- There is preservation of system redundancy, independence, and diversity.
- There is no over-reliance on programmatic or operator actions as compensatory measures.
- Common cause failures have been appropriately considered.
- The overall redundancy and diversity among the plant's systems and barriers is sufficient to ensure that no significant increase in risk would occur.

5.14.6 For Non-Safety Related but Important-to-Safety LSS components, **consider** if the risk information used in the categorization process provides an adequate basis for categorizing the SSC as LSS, in accordance with the following guidance:

- In general, the risk analyses should address the SSC function(s) that caused it to be originally classified as important-to-safety in order for an LSS categorization to be justified.
- <u>IF</u> the IDP concludes that the categorization of the SSC as LSS is <u>NOT</u> justified, <u>THEN</u> **consider** re-categorizing the SSC to HSS.
- <u>IF</u> the SSC is re-categorized from LSS to HSS, <u>THEN</u> **identify** the attributes of the SSC to assure that any core damage prevention and mitigation attributes that the IDP felt were significant are included in future treatment including beyond design basis functions used in the PRA.
-

5.15 { TC "5.15 IDP Overall Assessment " \f C \l "2" }IDP Overall Assessment

After evaluating the results in Section 5.14, **strive** to reach consensus on the overall categorization of the system functions and components, subject to the following:

- IF a component has been identified as HSS by the passive risk assessment, THEN **categorize** that component as HSS, regardless of any other factors.

NOTES

- For components that have both an active and a passive function, the overall risk of the component will be the higher of the two.
- It is important to continue to assess the active risk and the passive risk separately. For example, even though an active valve may be assessed as HSS due to its passive risk, the active risk should be separately determined.
- Typically, the PRA and qualitative risk assessments focus on the active risk. The separation of the two risks becomes useful when identifying component critical attributes in Section 5.16. The following criteria generally involve the active risk.

- IF a component has been identified as HSS by the PRA integrated risk assessment, THEN **categorize** that component as HSS, regardless of any other factors.
- IF a component has been identified as HSS by one or more of the non-modeled hazards evaluations, THEN **categorize** that component as HSS, regardless of any other factors.
- **Consider** revising the qualitative risk of system functions **OR** components from LSS to HSS based on IDP judgment. Conversely, **consider** revising the qualitative risk of components, in rare instances, from HSS to LSS IF an appropriate justification can be made, documented, and accepted by the IDP, subject to the guidance in Section 5.10.2.
- For components that are still LSS, **evaluate** the results of the sensitivity studies to determine if the component risk should be increased to HSS.
- For components that are still LSS, IF the results of defense-in-depth assessments point to a risk of HSS, THEN **revise** the risk to HSS UNLESS a justification can be made, documented, and accepted by the IDP that the risk should NOT be increased.
- For components that are still LSS, IF the component PRA results for CDF or LERF are in the PRA buffer zone (i.e., within 10% of the HSS threshold), THEN **consider** increasing the risk to HSS.

5.16 { TC "5.16 Component Critical Attribute " \f C \l "2" }Component Critical Attributes

5.16.1 For those components categorized as HSS, **identify** the attributes of the component that are associated with its safety significance. Typically, **develop** such attributes from one or more of the following sources:

- **Review** the HSS functions that the component supports and **determine** those actions that the component must perform in order to support the function(s).
- For PRA-modeled components, **examine** the associated failure mode (basic event) and **develop** the critical attribute as the opposite (e.g., "fail to start on demand" results in an attribute of "start on demand").

- For components that were assessed with a passive risk of HSS, **include** the critical attribute of pressure retention, as a minimum.

5.16.2 For those components supporting HSS functions but categorized as LSS based on mitigating factors, **identify** the attributes of the component that are associated with supporting the HSS functions as CRITICAL, with the clarification that loss of the attribute would not, in and of itself, fail the function.

5.17 { TC "5.17 Final Classification " \f C \l "2" }Final Classification

Classify the SSCs based on the combination of their safety significance and their safety related classification as follows:

RISC-1: SSCs that are safety-related and have been categorized as HSS

RISC-2: SSCs that are non-safety-related have been categorized as HSS

RISC-3: SSCs that are safety-related and have been categorized as LSS

RISC-4: SSCs that are non-safety-related and have been categorized as LSS

Document the results of the final classification of SSCs as detailed in Section 6.

5.18 { TC "5.18 Periodic Reviews " \f C \l "2" }Periodic Reviews and Performance Feedback

5.18.1 **Conduct** periodic reviews to ensure continued validity and performance monitoring for those SSCs that have been categorized. **Ensure** that the periodic reviews accomplish the following objectives:

- Are conducted at least once every two Unit 1 refueling outages
- Evaluate changes to the plant, operational practices, and applicable plant and industry operational experience for impact on existing categorizations
- Incorporate PRA model updates into the categorizations, including updated sensitivity studies results, as applicable
- Incorporate new PRA modeling capabilities
- Evaluate RISC-3 component performance since the last review to ensure that performance is acceptable and that no declining trends are noted. Specific attention should be focused on those components that have had alternative treatments applied to them.
- Evaluate RISC-2 component performance since the last review to ensure that no additional controls are needed to ensure that safety significant functions can still be supported.

5.18.2 IF significant changes to the plant risk profile are identified, OR IF it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent an HSS function from being satisfied, THEN **perform** an immediate evaluation and review prior to the normally scheduled periodic review.

5.18.3 IF a change to the categorization of an SSC is suggested either by: 1) a change in plant design or operation that would prevent a safety-significant function from being satisfied OR 2) by a change in the PRA model as determined from the absolute importance measures, THEN **present** such changes to the IDP for concurrence, in accordance with the following:

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

NOTE

Risk insights from new PRA models (e.g., seismic model) do not necessarily require a re-categorization of the system, unless such insights point to a higher integrated risk than the current overall risk of the component(s). In such cases, only the affected components need to be evaluated for potential re-categorization.

5.18.4 IDP Review – **Convene** to review the results of these reviews and **determine** if any of the following features require revision:

- Risk of system functions and/or components
- Alternative treatments being currently applied
- Component critical attributes
- Documented categorization basis

NOTE

The IDP has the final decision regarding re-categorizations.

5.19 { TC "5.19 Critical Changes " \f C \l "2" }Critical Changes

NOTES

- As allowed by 10 CFR 50.69, RISC- 3 components can be removed from the scope of many special treatment requirements and subjected to alternative treatment requirements. A change to the categorization of a RISC-3 component from LSS to HSS will result in the component being classified as RISC-1. This type of change is considered a critical change and is to be addressed expeditiously. Critical changes apply to safety-related components only.
- A critical change occurs whenever the risk of a <u>safety-related</u> component changes from LSS to HSS. Components that have <u>NOT</u> had any alternative treatments applied are <u>NOT</u> subject to critical changes. Critical changes do <u>NOT</u> apply to increases in the risk of system functions; however, such changes can result in a critical change at the component level.
- Critical changes are most likely to occur following a revision to the PRA Model(s). However, critical changes may also occur due to new insights, negative performance trends, design changes, etc.

5.19.1 As soon as the potential for a critical change is identified, **initiate** a Condition Report in accordance with the Corrective Action Program. **Ensure** that the Condition Report includes the necessary data to support a proper evaluation and contains, at a minimum, the following actions:

NOTE

<u>IF</u> conditions/events do <u>NOT</u> permit the below timeframes to be satisfied, <u>THEN</u> the IDP Chair <u>SHALL</u> ensure that interim compensatory measures are instituted until the next required action can be accomplished.

- **Convene** the IDP to determine the appropriateness of the potential change within 14 calendar days of the initiation of the Condition Report action.
- <u>IF</u> an electronic database is being used to provide RISC classifications for use by the Plant, <u>THEN</u> **revise** the database to reflect the new RISC-1 classification within 14 days of IDP approval of the change.
- **Amend** the Risk Basis Document for the applicable system to reflect the new RISC-1 classification within 30 days of IDP approval of the change.
- **Perform** an evaluation to determine the acceptability of activities performed on, or for, the component during the time that the component was under the RISC-3 classification. **Consider** license compliance and operability as necessary.
- Within 10 calendar days of IDP approval of the change, **notify** the owner of each alternative treatment program that may be impacted by the change. **Assign** individual actions to each owner to complete the assessment.

5.19.2 In the unlikely event that the IDP decides that the critical change is <u>NOT</u> valid, **notify** the owners of the associated Condition Report action items as soon as possible, **revise** the action items to incorporate the decision of the IDP, and **rescind** any changes, as applicable.

6.0 { TC "6.0 Records" \f C \l "1" }Records

> **NOTE**
>
> The development and evaluation of risk insights that support the categorization of SSCs as detailed in this instruction as well as in the associated instructions (NMP-ES-065-001 and NMP-ES-065-002) and procedure (NMP-ES-066) SHALL be documented to ensure that the process and results are scrutable, consistent, and reflect the current plant design.

6.1 Ensure that the following data or information is documented:

- Procedures, instructions, or guidelines that describe the processes for the development, evaluation, and use of the SSC categorizations
- System functions – identified and categorized with the associated bases
- Mapping of components to supported function(s)
- PRA model results, including sensitivity studies
- Hazards analyses, as applicable
- Passive risk assessment results and bases
- Categorization results for components, including ALL associated bases and the RISC classifications
- Component critical attributes
- Results of periodic reviews and SSC performance evaluations
- IDP meeting minutes with associated attachments

6.2 **Identify** the above documents as QA records and **store** them in the **Corporate doc base** using the R type identified below.

6.2.1 After the IDP approves categorization results of a system, **document** the results in a Risk Based Document (RBD). **Ensure** that the RBD contains associated supporting information that was used to categorize the system. **Store** the RBD in the Corporate doc base and **assign** it the Corporate R Type of PRA05.017.

6.2.2 **Store** the IDP meeting minutes per NMP-ES-066-001.

6.3 **Consider** the use of a suitable plant-wide electronic means of providing the RISC classifications of components. **Update** this data to reflect categorization data changes within a reasonable period of time, notwithstanding the specific time constraints associated with critical changes.

6.4 **Update** the RBD to incorporate changes to categorization data, if applicable, at least at the same frequency as the scheduled Periodic Review for the associated system. **Implement** this update through a general revision to the RBD that incorporates any changes to the categorization data identified since the last revision, including those identified during the Periodic Review process.

6.5 As an option, **consider** updating the RBD through the use of an amendment-type change process. **Incorporate** outstanding amendments through a general revision on at least the same frequency as the scheduled Periodic Review for the associated system.

7.0 { TC "7.0 References" \f C \l "1" }References

7.1 10 CFR 50.69, Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors

7.2 NEI 00-04, 10 CFR 50.69 SSC Categorization Guide, Revision 0

7.3 NMP-ES-065, 10 CFR 50.69 Program

7.4 NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights

7.5 NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization

7.6 NMP-ES-066: General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

7.7 NMP-ES-066-002: Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

8.0 { TC "8.0 Commitments" \f C \l "1" }Commitments

None

{ TC " Attachment 1 – Guidelines for Defense-in-Depth Assessments " \f C \l "1" }Attachment 1 – Guidelines for Defense-in-Depth Assessments

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

1. Core Damage Defense-in-Depth

The initial assessment should consider both the level of defense-in-depth in preventing core damage and to the frequency of the events being mitigated. Figure 1 is an example of such an assessment. This figure depicts the internally initiated design basis events considered in the plant's safety analysis report (i.e., the events that were used to identify an SSC as safety-related) and considers the level of defense-in-depth available, based on the success criteria used in the PRA. This ensures that adequate defense-in-depth is available to mitigate design basis events. The defense-in-depth matrix is similar in form to the Significance Determination Process used in the Reactor Oversight Process and uses the same concepts of diverse and redundant trains and systems in evaluating the level of defense-in-depth.

The following process is used in applying Figure 1. For each active component/function categorized as LSS:

- **Identify** the design basis events for which the function is required.
- For each design basis event, **identify** the other systems and trains that can support the function OR can provide an alternative success path to avoid core damage. Potential combinations of other systems and trains are depicted across the top row of Figure 1. Credit may be taken for systems containing RISC-1, 2, 3, or 4 SSCs (with the exception noted in the bullet below), and realistic success paths may be used.
- For each design basis event, **identify** the region of Figure 1 in which the plant mitigation capability lies WITHOUT credit for the function/SSC that has been proposed as low safety-significant, and WITHOUT credit for any identical, redundant SSCs within the system that are also classified as LSS.
- IF the result is in the region entitled "Low Safety Significance Confirmed," THEN **retain** the categorization of the function/SSC.
- IF the result is in the region entitled "Potentially Safety-significant," THEN **classify** the function/SSC as HSS for the IDP, noting that the basis is core damage defense-in-depth.

When complete, IF ALL SSC functions are confirmed as LSS, THEN **retain** the SSC classification as candidate LSS for the IDP.

Examples:

- For a BWR, if the low pressure core spray (LPCS) system pumps were LSS in the categorization process using risk information, then their categorization would be confirmed using Figure 1. In this case, the LPCS pumps have the function of providing coolant makeup to the RPV at low pressure. This function is required either (a) in response to a large LOCA, or (b) in response to other transients and LOCAs where other coolant makeup systems are failed.
- For mitigation of a large LOCA, the low pressure coolant injection (LPCI) function of the RHR system can also support the coolant inventory makeup function. The LPCI function is automatic and consists of at least two redundant trains. Thus, for this LOCA event, in the bottom row of

Figure 1, the presence LPCI as a redundant automatic system confirms the low safety significance of LPCS.

- In order to confirm LSS in high frequency transient events, such as reactor trip, either two redundant systems are required or three or more trains must exist. For BWRs, there are multiple coolant inventory makeup systems that could be used without crediting LPCS (i.e., HPCI, Reactor Core Isolation Cooling (RCIC), main feedwater, condensate, and LPCI with Automatic Depressurization System (ADS)). This exceeds the redundancy and diversity requirements for mitigation of these events.
- In order to confirm LSS for mitigation of a stuck open relief valve, one train plus one redundant system is required. In this case, BWRs have LPCI with ADS and HPCI plus control rod drive cooling (CRD) to provide success paths. This provides a redundant system (LPCI/ADS) and one additional diverse train (HPCI/CRD).
- In order to confirm LSS for mitigation of loss of one safety-related DC bus, at least two diverse trains are required. In this case, BWRs would have one train of LCPI and either HPCI (a one train system) or RCIC (a one train system) available to meet the requirement for two diverse trains

2. Containment Defense-in-Depth

Defense-in-depth should also be assessed for SSCs that play a role in preventing large, early releases. Level 2 PRAs have identified the several containment challenges that are important to LERF. These include containment bypass events such as ISLOCA (BWR and PWR) and SGTR (PWR), containment isolation failures (BWR and PWR), and early hydrogen burns (ice condensers and Mark III). Containment defense-in-depth is also assessed for SSCs that play a role in preventing large containment failures (e.g., due to loss of containment heat removal). For each SSC function categorized as candidate LSS, its defense-in-depth is assessed using the following criteria:

<u>Containment Bypass</u>

- Can the SSC initiate an ISLOCA event?
- Can the SSC provide a significant level of mitigation of an ISLOCA event?
  [Note that mitigation (up to and including isolation) of ISLOCA is a beyond design basis function. There are a number of SSCs that could be credited with providing varying degrees of mitigation of an ISLOCA. However, only SSCs providing a significant level of mitigation should be candidate HSS. These SSCs would also be treated in the internal events model as LERF mitigators, and thus their significance would be considered in that aspect of the categorization process.]
- Can the SSC isolate a faulted steam generator following a steam generator tube rupture event?

<u>Containment Isolation</u>

- Does the SSC support containment isolation for containment penetrations that are:
  - Directly connected to containment atmosphere, and
  - > 2" in diameter, and
  - <u>NOT</u> locked closed <u>OR</u> only locally operated?
- Does the SSC support containment isolation for containment penetrations that are:
  - Part of the reactor coolant system pressure boundary, and
  - > 3/8" in diameter, and
  - <u>NOT</u> locked closed <u>OR</u> only locally operated?

Early Hydrogen Burns

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

Long-Term Containment Integrity

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

In cases where the answer to any of the above questions is "yes," the SSC should be categorized as candidate HSS. IF ALL of the above questions are answered "NO," THEN LSS is confirmed. When complete, IF ALL SSC functions are confirmed as LSS, THEN **retain** the SSC classification as candidate LSS for the IDP.

In cases where SSCs are identified as HSS, the safety-significant attributes should be defined. This involves identifying the performance aspects and failure modes of the SSC that contribute to it being safety-significant. These attributes are to be provided to the IDP.

Attachment 1, Figure 1

DEFENSE-IN-DEPTH MATRIX

Frequency	**Design Basis Event**	≥3 diverse trains <u>OR</u> 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^{3}yr	SGTR Stuck Open PORV/SV RCP Seal LOCA MFLB MSLB Inside Loss of 1 SR DC bus	**LOW SAFETY SIGNIFICANCE CONFIRMED**			
<1 per 10^3 yr	LOCAs Other Design Basis Accidents				

Instruction Owner: Paul Hayes / Fleet Engineering Services Director / Corporate
(Print: Name / Title / Site)

Approved By: Original signed by Paul Hayes / 11/22/2011
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:	11/23/2011	N/A	N/A	11/23/2011	N/A
	Corporate	FNP	HNP	VEGP 1-2	VEGP 3-4

PRB Required
This NMP is under the oversight of the Risk-Informed Engineering Department

Writer(s):
Vish Patel
Stephanie Agee

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Version Number	Version Description
1.0	Initial Issue

Table of Contents

1.0 Purpose

1.1 This instruction describes the Integrated Decision-making Panel (IDP) in its role as a multi-disciplinary review group for the 10 CFR 50.69 risk informed categorization process. This instruction defines the IDP's structure, responsibilities, and qualifications.

> **NOTE**
>
> Throughout this instruction, reference to the IDP refers only to the IDP's role in supporting the 10 CFR 50.69 risk informed categorization process. Other risk-informed programs may utilize an IDP whose attributes (e.g., role, composition, qualifications, etc.) are not in the scope of this instruction.

1.2 This instruction is part of an integrated categorization process which includes the following additional procedures/instructions:

- NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
- NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization
- NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components
- NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

1.3 The process described in this instruction is considered to satisfy the requirements of 10 CFR 50.69 paragraph (c)(2), *SSC Categorization Process*, and partially satisfy paragraph (e), *Feedback and Process Adjustment*, and paragraph (f), *Program Documentation, Change Control, and Records*. The scope of this instruction does NOT include alternative treatment requirements specified in 10 CFR 50.69 (d).

2.0 Applicability

This instruction applies to personnel involved in the integrated decision-making process for the 10 CFR 50.69 program.

3.0 Definitions

3.1 **Alternate** – An individual selected by the IDP Chairperson to serve in the absence of a primary member. Each alternate SHALL meet the minimum qualifications for the IDP member that the alternate is replacing.

3.2 **Consensus** – a group decision making process that not only seeks the agreement of most participants, but also the resolution of differing opinions or objections. The process does not involve a simple vote, but also consideration of relevant issues raised by the members of the group. For purposes of the IDP, agreement on an outcome by at least a two-thirds majority of the quorum members is considered consensus. Consensus is required for final decisions regarding the safety significance of Structures, Systems, and Components (SSCs).

4.0 Responsibilities

> **NOTE**
>
> Detailed guidance for IDP responsibilities is provided in NMP-ES-065-003.

4.1 Site IDP

- Evaluates PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization for system functions and components.
- Reviews results from performance monitoring and periodic reassessments to ensure that the basis for the categorization of SSCs remains valid and that any implemented alternative treatments have <u>NOT</u> significantly degraded the performance of the associated components.
- Evaluates recommended changes to categorization resulting from changes to the plant, PRA model updates, changes to operational practices, as well as other applicable changes.

4.2 Site IDP chairperson

- Schedules and runs the site IDP meetings.
- Ensures that quorum requirements are met for IDP meetings.
- Ensures site IDP meeting minutes are prepared.
- Ensures site IDP meeting minutes are approved.

4.3 Risk Informed Engineering Department

- Ensures that minutes of site IDP meetings are retained along with other required IDP records per site QA records process.
- Notifies the site IDP Chairperson when a 50.69 IDP meeting is needed.
- Ensures that training is developed for the IDP
- Provides qualification training.

5.0 Procedure

NOTES

- This procedure has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Activities described in this procedure may be performed prior to NRC approval of the license amendment.
- The alternative treatment requirements specified in 10 CFR 50.69 (d) SHALL NOT be implemented UNLESS step 5.1, step 5.2, and, if applicable, step 5.3 are verified to be complete.

5.1 After the license amendment is approved by the NRC, **perform** a documented evaluation to ensure that the process described in this procedure meets the requirements of, and is consistent with, the NRC-approved license amendment.

5.2 **Track** the performance of this evaluation via a Condition Report action. This evaluation SHALL be approved by the Director, Risk-Informed Engineering and by the Director, Licensing. After approval of the evaluation, **revise** this procedure to remove the above Note, this step and steps 5.1 and 5.3.

5.3 IF the above evaluation concludes that the process described in this procedure DOES NOT meet the requirements of, or is inconsistent with, the approved license amendment, THEN **revise** this procedure accordingly and **re-perform** any evaluations or activities already performed using the revised procedural requirements.

NOTE

Detailed guidance for IDP responsibilities is provided in NMP-ES-065-003.

5.4 IDP Organization

5.4.1 Composition and Quorum

5.4.1.1 The site IDP should include members from the following organizations:

- Site Operations (SRO)
- Safety Analysis
- Site Design Engineering
- Site System Engineering
- Site Risk Informed Application
- Site Nuclear Licensing
- Site Maintenance

5.4.1.2 A Quorum for the IDP SHALL consist of at least five qualified persons, collectively having site specific expertise in the following functional areas:

- Operations (SRO)
- Safety Analysis
- Design Engineering
- Systems Engineering
- Probabilistic Risk Analysis (PRA)

5.4.1.3 The Operations Manager (OR designee) selects primary and alternate members to serve on the IDP.

- The qualified alternate(s) are designated to sit on the panel for absent member(s).
- The Operations Manager (OR designee) will act as a Chairperson.

5.4.2 Qualifications

5.4.2.1 All members SHALL have:

- Understanding of PRA concepts and the analyses performed for risk informed categorization.
- Understanding of the risk informed categorization process.
- Understanding of the risk informed categorization requirements.
- Experience with the specific plant being evaluated.

5.4.2.2 All IDP members SHALL have completed an IDP member qualification form.

5.4.2.3 For each functional area, **consider** having an alternate member complete the qualification process so as to substitute for the primary member, if needed.

5.4.3 IDP Training

5.4.3.1 Initial IDP Training for the IDP SHALL include:

- The purpose of risk informed categorization including exempted regulations for low safety significance SSCs.
- The categorization process
- Risk informed defense in depth philosophy and how it is maintained.
- Details of the IDP process including roles and responsibilities
- PRA fundamentals pertinent to the 50.69 program
- Details of the specific plant PRA analyses used for the preliminary categorization including:
  - model scope and assumptions (ALL hazard groups)
  - interpretation of risk importance measures
  - role of sensitivity studies and changes in risk evaluations (e.g., impact of PRA model updates or additional PRA models)

5.4.3.2 Initial training SHALL be documented using the form NMP-ES-066-002-F01, as detailed below.

- The qualifications are listed in the above form for each member.
- The qualifications (#-IDP-QL-###) are site specific. Therefore, substitute the first letter (#) in the qualification for a site. For example, qualification for Vogtle IDP chairperson would be "*V*-IDP-QL-*Chairperson*".

5.4.3.3 Refresher training should be provided to IDP members every 3 years. Refresher training will be conducted via CBT (or equivalent) and linked to qualifications listed in form NMP-ES-066-002-F01.

5.5 IDP Activities

5.5.1 Meetings

5.5.1.1 The IDP SHALL meet when ANY of the following apply:

- When a risk categorization is completed in accordance with NMP-ES-065-003 and ready for IDP review.
- When plant OR PRA changes require re-evaluation of categorization results.
- To review the results of the periodic review of the program.
- As convened by the chairperson.

5.5.1.2 Meetings SHALL NOT be conducted without a quorum present.

5.5.1.3 For scheduled IDP meetings, **strive** to have ALL primary members present. If a primary member's absence is unavoidable, an alternate may be called. The primary member should notify the Chairperson in advance of the meeting, if practical, stating the reason(s) for the absence.

5.5.2 Minutes of Meetings

5.5.2.1 The IDP Chairperson will ensure the minutes of IDP meetings are prepared.

5.5.2.2 At a minimum, the minutes SHALL include:

- The quorum members attending the meeting,
- Verification that there was a quorum present,
- The meeting agenda,
- The results of the IDP activities including the outcome of the categorization review, the basis for the determination, ANY differing opinions, and ANY significant issues discussed leading to the decision,
- Open actions from the meeting.

NOTE

Refer to NMP-ES-066-002-F02 for an example of an acceptable format for meeting minutes.

5.5.2.3 **Number** the minutes sequentially for each calendar year.

5.5.2.4 **Provide** the prepared minutes to the IDP members for review. **Ensure** that the minutes are approved by the Chairperson (or alternate). Approval should be within 30 days of the meeting.

5.5.2.5 **Retain** the meeting minutes as a quality record. **Ensure** that the meeting minutes are stored per site QA records process. Fleet R Type is RR5.018.

6.0 Records

This instruction, and any documents created using this instruction, will become QA Records when completed unless otherwise stated. The associated documents are considered complete when issued into DMS.

QA record (X)	Non-QA record (X)	Record Generated	Retention Time	R-Type*
X		Risk Informed Categorization Integrated Decision Making Panel Qualification Form -50.69	Life of Plant	TR0.001
X		Risk Informed Categorization Integrated Decision Making Panel Meeting Minute Form -50.69	Life of Plant	RR5.018

* The R type is for fleet.

7.0 References

7.1 NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline", Revision 0, July 2005.

7.2 R.G. 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" Revision 1, July 2006.

7.3 NMP-ES-065, 10 CFR 50.69 Program

7.4 NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components

7.5 NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program

8.0 Commitments

None

Risk Informed Categorization Integrated Decision Making Panel (IDP)

Training/ Qualification Record for ______________________(site); LMS Qualification ID: #-IDP-QL-###

Last Name First Name MI

Part A - The following documents shall be read and studied to the extent necessary to obtain a working knowledge of the administrative processes and requirements, preferably prior to completing the training in Part B:

1. Risk informed categorization procedures:
   NMP-ES-065: 10 CFR 50.69 Program
   NMP-ES-065-001: 10 CFR 50.69 Active Component Risk Significance Insights Instruction
   NMP-ES-065-002: 10 CFR 50.69 Passive Components Categorization Instruction
   NMP-ES-65-003: 10 CFR 50.69 Risk Informed Categorization for Systems, Structures, and Components
   NMP-ES-065-004: Alternate Treatment Requirements
   NMP-ES-066: General Guidance for Decision Making Panels - 50.69 and Surveillance Frequency Control Program
   NMP-ES-66-002: Integrated Decision-Making Panel for Risk Informed SSC Categorization: Duties And Responsibilities
2. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline", Revision 0, July 2005.
3. R.G. 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" Revision 1, July 2006.

CERTIFICATION THAT ABOVE READING IS COMPLETE:

(Signature)____Date:________________.

Part B – Risk informed categorization training session completed:

Date:

Part C - Personal Data Summary:

1. Expertise area (# = F, H, V):

[ ] IDP Chairperson [#-IDP-QL-Chairperson]

[ ] Plant Operations [#-IDP-QL-Operations]

[ ] Design Engineering [#-IDP-QL-Design Engineering]

[ ] System Engineering [#-IDP-QL-System Engineering]

[ ] Probabilistic Safety Assessment [#-IDP-QL-PRA]

[ ] Safety Analysis [#-IDP-QL-Safety Analysis]

[ ] Licensing [#-IDP-QL-Licensing]

[ ] Maintenance [#-IDP-QL-Maintenance]

Risk Informed Categorization Integrated Decision Making Panel (IDP)

Training/ Qualification Record for ____________________(site); LMS Qualification ID: #-IDP-QL-###

Last Name	First Name	MI

4. Document Industry Experience in above area(s):

5. Document Plant Specific Experience:

6. Other Specific Area(s) of expertise and experience:

Part D – Approval

Line Organization Acknowledgement of the IDP Responsibilities

The individual listed on this form will represent the organization/expertise area identified below. Sufficient resources will be provided to perform the IDP roles and responsibilities in NMP-ES-066.

Organization/expertise Represented:

[ ] IDP Chairperson [#-IDP-QL-Chairperson]

[ ] Plant Operations [#-IDP-QL-Operations]

[ ] Design Engineering [#-IDP-QL-Design Engineering]

[ ] System Engineering [#-IDP-QL-System Engineering]

[ ] Probabilistic Safety Assessment [#-IDP-QL-PRA]

[ ] Safety Analysis [#-IDP-QL-Safety Analysis]

[ ] Licensing [#-IDP-QL-Licensing]

[ ] Maintenance [#-IDP-QL-Maintenance]

Manager of Department Represented: ______________________________ Date: ________

Site IDP Chairperson: ______________________________ Date: ________

Risk Informed Categorization Integrated Decision Making Panel (IDP)

Training/ Qualification Record for ______________________(site); LMS Qualification ID: #-IDP-QL-###

Last Name	First Name	MI

Part E

Once this form has been completed, it shall be forwarded to Training Supervision to update the LMS by giving credit to LMS Qual ID #-IDP-QL-###.

____

Training Supervision   Date

NOTE

Training Supervision shall process this record to the Document Management System in accordance with NMP-TR-112.

50.69 Integrated Decision-making Panel (IDP) Meeting

Month Day, Year

Meeting Number: YY-##

Minutes Approved: ______________________
IDP Chair

Purpose: **10 CFR 50.69 Categorization of the XXXX system**

Attended ☐ Quorum met

☐ ________________,Chair, Operations
☐ ______________, System Engineering
☐ ________________, Safety Analysis
☐ ________________, Nuclear Licensing
☐ ___________, Site Design Engineering
☐ ________________, PRA

Others Present:

Mr. (Chairman) called the meeting to order and began with a discussion on Target Zero, INPO nuclear safety principle of the week, and operating experience. (Any additional message)

Mr. (Chairman) called the meeting to order. The agenda and desired outcomes were reviewed. The XXXX System Categorization package (Attachment 1) was presented to the IDP. The following is a summary of the discussions and relevant comments or actions.

Editorial Changes

Package Section	Comment	Response or Follow-up Action

Non-Editorial Comments/Actions:

No.	Description	Responsibility
1.		
2.		
3.		

The meeting was adjourned with the following plusses and deltas:

Plusses	Deltas

Attachments:

QA record (X)	Non-QA record (X)	Record Generated	Retention Time	R-Type
X		Meeting Minutes for Risk Informed Categorization IDP (50.69)	Life of Plant	RR5.018

Instruction Owner: Paul Hayes / Fleet Engineering Services Director / Corporate
(Print: Name / Title / Site)

Approved by: Original signed by R. Lee Mansfield for Paul Hayes on 01/18/2013
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:

01/23/2013	01/23/2013	01/23/2013	01/23/2013	N/A
Corporate	FNP	HNP	VEGP 1&2	VEGP 3&4

PRB Review Required

Writer(s):
Vish Patel
Stephanie Agee

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Version Number	Version Description
1.0	Initial Issue
1.1	Making of procedure effective for plant Hatch
2.0	Deleted NMP-ES-066-001 and NMP-ES-066-001-F01. This procedure was updated to remove all references except a note in the reference section that NMP-ES-066-001 was deleted. All IDP guidelines for the SFCP are now located in NMP-ES-072-006. NMP-ES-066-001-F01 has been replaced by NMP-ES-072-006-F01.

Table of Contents

1.0 Purpose

This procedure establishes the concepts of the Integrated Decision-making Panel (IDP) for the 50.69 (Risk Informed Categorization (RIC)) process (for which specifics are described in NMP-ES-066-002). The process specific Site IDPs approve the results of the 50.69 process.

2.0 Applicability

This procedure is applicable to the 50.69 process with regard to their use of IDPs.

3.0 References

3.1 NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline", Revision 0, July 2005.

3.2 R.G. 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" Revision 1, July 2006.

3.3 NMP-ES-065, 10CFR50.69 Program.

3.4 NMP-ES-066-001 (Deleted).

3.5 NMP-ES-066-002, Integrated Decision-Making Panel for Risk Informed SSC Categorization Duties and Responsibilities.

3.6 NMP-ES-066-001-F02 – IDP Meeting Minutes.

4.0 Definitions

4.1 Integrated Decision-making Panel (IDP) – A multi-disciplinary panel of plant – knowledgeable experts that considers both risk and deterministic inputs to determine whether a proposed plant change is appropriate; considering plant design and operating practices and experience in addition to risk insights.

4.1.1 50.69 IDP- the IDP convened to review risk informed categorization of structures systems and components.

4.2 Consensus – a group decision making process that not only seeks the agreement of most participants, but also the resolution of differing opinions or objections. That is, not a simple vote, but also consideration of relevant issues raised by the members of the group. For purposes of the IDP, agreement on an outcome by a two-thirds majority of the quorum members is considered consensus. Consensus is required for final decisions regarding safety significant and LSS.

5.0 Responsibilities

5.1 An IDP has the following responsibilities.

5.1.1 Serve as a multi-disciplinary review panel collectively having broad knowledge of plant design, licensing requirements, operating and maintenance practices, risk and experience.

5.1.2 Ensure all attributes of the evaluation presented to them are fully addressed to provide a valid risk informed conclusion or decision that addresses the maintenance of defense-in-depth and adequate safety margin.

5.2 The responsibilities of the site IDP for the 10CFR 50.69 Categorization Process and the IDP Chairperson are defined in NMP-ES-066-002.

5.3 Risk Informed Engineering Department.

5.3.1 Ensures that training is developed for the IDPs as required.

5.3.2 Ensures IDP members have the appropriate training and/or qualifications before participating in IDP deliberations.

5.4 Site Operations Manager or designee.

5.4.1 Selects individuals to serve as IDP members.

5.4.2 Serve as IDP chairperson.

6.0 Procedure

6.1 Site IDP.

6.1.1 A site IDP is composed of members of varying disciplines as defined by the applicable guidance document for the specific process (e.g. 10CFR50.69 or Surveillance Frequency Control Program).

6.1.2 IDP members are required to be qualified for participation in the IDP.

6.1.3 The site IDP is envisioned as a group that meets the requirements of the 50.69 processes. Depending on which process convenes the IDP, the quorum requirements will vary. The IDP chairperson ensures that the appropriate quorum requirements are met.

6.1.4 The site Operations Manager (or designee) selects individuals to serve on the site IDP, with concurrence of the individuals' department manager.

6.1.5 The site Operations Manager (or designee) will act as Chairperson.

6.1.6 The site IDPs will meet on an as needed basis or as designated in the process specific procedures.

6.1.7 A site IDP shall be convened to review material related to a single process.

6.1.8 The IDP reviews the material presented to it and makes a decision whether to recommended HSS/LSS categorization. The decision should be a consensus.

6.1.9 The material should be discussed until the consensus is achieved. The IDP Chairperson should ensure discussion is not limited or dominated by any one member.

6.1.10 If there is a dissenting opinion that is not easily resolved by additional information or review, the dissenting opinion and issue must be documented in the IDP minutes. This should be a rare occurrence.

6.1.11 IDP meeting minutes shall be documented and retained within the records management system.

7.0 Records

7.2 Records related to the 10 CFR 50.69 Categorization Process are defined in NMP-ES-066-002.

8.0 Commitments

None.