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SOUTHERN COMPANY

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
Draft Risk-Informed Categorization Procedures

Ladies and Gentlemen:

The Nuclear Regulatory Commission (NRC) by letter dated June 17, 2011, in response to Southern Nuclear Operating Company's (SNC) letter dated December 6, 2010, granted pilot status for the planned SNC Vogtle Electric Generating Plant (VEGP) 10 CFR 50.69 license amendment request.

On March 29, 2011, NRC and SNC met to review SNC's planned approach for implementation of 10 CFR 50.69, risk-informed categorization and treatment of structures, systems, and components (SSCs) for nuclear power reactors. SNC discussed the development of draft risk-informed categorization procedures implementing applicable NRC and industry guidance, specifically NRC Regulatory Guide (RG) 1.201 Revision 1 and NEI 00-04 Revision 0 which is endorsed by RG 1.201. The draft categorization procedures are being used during the ongoing trial categorization of three VEGP systems to test the efficacy of the categorization process prior to documenting the process in the VEGP 10 CFR 50.69 license amendment request.

In response to an NRC request at the referenced meeting, this letter provides the draft risk-informed categorization procedures in Enclosures 1–7.

This letter contains no NRC commitments. If you have any questions, please contact Jack Stringfellow at (205) 992-7037.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

MJA/CLT/lac

Enclosures:
1. Draft NMP-ES-065, 10 CFR 50.69 Program
2. Draft NMP-ES-065-001, 10 CFR 50.69 Active Component Risk Significance Insights
3. Draft NMP-ES-065-002, 10 CFR 50.69 Passive Component Categorization
4. Draft NMP-ES-065-003, 10 CFR 50.69 Risk Informed Categorization for Structures, Systems, and Components
5. Draft NMP-ES-066, General Guidance for Decision-Making Panels – 50.69 and Surveillance Frequency Control Program
6. Draft NMP-ES-066-002, Integrated Decision-Making Panel for Risk Informed SSC Categorization: Duties and Responsibilities
7. Draft NMP-ES-066-002-F01, Risk Informed Categorization Integrated Decision Making Panel Qualification Form – 50.69

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. J. T. Gasser, Executive Vice President
Mr. T. E. Tynan, Vice President – Vogtle
Ms. P. M. Marino, Vice President – Engineering
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U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
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Mr. L. M. Cain, Senior Resident Inspector – Vogtle

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 1

Draft NMP-ES-065
10 CFR 50.69 Program

Procedure Owner: ______________________________

(Print: Name / Title / Site)

Approved By: ______________________________

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

Effective Dates: ________ Corporate ________ FNP ________ HNP ________ VEGP 1-2 ________ VEGP 3-4

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

Writer(s):

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

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on site for reference as needed.	**ALL**

Revision Description

Version Number	Revision Description
1.0	Initial issue

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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 and treatment of SSCs.

- NMP-ES-065-001, 10CFR50.69 Active Component Risk Significance Insights
- NMP-ES-065-002, Passive Component Categorization
- NMP-ES-065-003, Risk Significance Categorization for Systems, Structures, and Components
- NMP-ES-065-004, Alternative Treatment Requirements (To be developed)
- NMP-ES-066, Integrated Decision Making Panel (IDP)
- NMP-NL-XXX, Nuclear Licensing Procedure for Implementation of 10 CFR 50.69

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

1.5 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. However, the alternative treatment requirements specified in 10 CFR 50.69 (d) shall **NOT** be implemented **UNLESS** the following actions are verified to be completed:

1.5.1 After the license amendment is approved by the NRC, an evaluation shall be performed and documented to ensure that the process described in this procedure meets the requirements of, and is consistent with, the NRC-approved license amendment. The performance of this evaluation shall be tracked via a Condition Report action. This evaluation shall be approved by the Manager, Risk-Informed Engineering and by the Manager, Licensing. The procedure shall then be revised at this time to remove this Section.

1.5.2 **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** this procedure shall be revised accordingly and any evaluations or activities already performed shall be re-performed using the revised procedural requirements.

2.0 Applicability

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, **ALL** the components in that system **MUST** be included in the categorization process.

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.

This procedure was created and is maintained under the direction of the Risk-Informed Engineering Manager.

3.0 References

3.1 10 CFR 50.69, Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors

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

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

3.4 NMP-ES-065-001, 10CFR50.69 Active Component Risk Significance Insights

3.5 NMP-ES-065-002, 10CFR50.69 Passive Component Categorization

3.6 NMP-ES-065-003, 10CFR50.69 Risk Informed Categorization for Systems, Structures, and Components

3.7 NMP-ES-065-004, Alternative Treatment Requirements

3.8 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

3.9 NMP-ES-066-001: Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

3.10 NMP-NL-XXX, Nuclear Licensing Procedure for Implementation of 10 CFR 50.69

4.0 Definitions

4.1 **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).

4.2 **Basic Safety Function (a.k.a Key Safety Function)** – one of the key safety functions of the plant, namely reactivity control, core cooling, heat sink, RCS inventory, and containment barrier (It is noted that loss of a single train would typically not constitute a loss of a function).

4.3 **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) to the affected system or equipment train.

4.4 **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).

4.5 **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)).

4.6 **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).

4.7 **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).

4.8 **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.

4.9 **Core Damage** – uncovery 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.

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

4.11 **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 Reg 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.

4.12 **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.

4.13 **Failure Mode** – 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, leaks)

4.14 **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.

4.15 **Failure Potential** – 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).

4.16 **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".

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

4.18 **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.

4.19 **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 shall include PRA, safety analysis, plant operations, design engineering, and system engineering.

4.20 **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.

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

4.22 **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.

4.23 **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.

4.24 **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

4.25 **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 of the failure.

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

4.27 **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)

4.28 **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 of an accident

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

4.30 **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).

4.31 **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 of an accident.

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

4.33 **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.

4.34 **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.

4.35 **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.

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

4.37 **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.

4.38 **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.

4.39 **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).

4.40 **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]

4.41 **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 human error rates, removal of common cause failures, increasing maintenance unavailability, and increasing the failure rate of LSS components.

4.42 **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.

4.43 **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 design considerations, qualification, change control, documentation, reporting, maintenance, testing, surveillance, and quality assurance requirements.

4.44 **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.

4.45 **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.

4.46 **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.

4.47 **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.

4.48 **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.

4.49 **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.

5.0 <u>Responsibilities</u>

5.1 The Manager, Risk-Informed Engineering is responsible for the following activities:

5.1.1 Managing the 10 CFR 50.69 Program

5.1.2 Ensuring PRA technical adequacy as required to support the 10 CFR 50.69 process

5.1.3 Assigning Risk-Informed Application Engineer(s) as required to support the Program

5.1.4 Providing training to IDP members and other selected site personnel

5.2 The IDP is responsible for the following activities:

5.2.1 Evaluating PRA risk insights, passive risk insights, and deterministic risk insights to reach a consensus-based categorization for system functions and components that are presented to the IDP for review.

5.2.2 Reviewing 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.

5.2.3 Evaluating recommended changes to categorization results resulting from changes to the plant, PRA model updates, changes to operational practices, as well as other applicable changes.

5.3 The cognizant Risk-Informed Application engineer is responsible for the following activities:

5.3.1 Providing PRA insights in support of the active risk categorization of system functions and components.

5.3.2 Providing PRA insights in support of the passive risk categorization of system components.

5.3.3 Providing the results of other hazards analyses for those hazards that are not modeled in the PRA.

5.4 The cognizant System Engineer is responsible for the following activities:

5.4.1 Developing system functions.

5.4.2 Mapping each component in the system to the system function(s) supported.

5.4.3 Participating in the categorization of active risk for system functions and components.

5.4.4 Participating in the categorization of passive risk for system components.

5.5 The Operations representative is responsible for the following activities:

5.5.1 Providing deterministic responses to the essential questions used to assess the risk of system functions.

5.5.2 Participating in the categorization of active risk for system functions and components.

5.5.3 Participating in the categorization of passive risk for system components.

5.6 The treatment program owner is responsible for the following within the scope of the program:

5.6.1 Evaluating alternative treatment options for RISC-3 SSCs

5.6.2 Evaluating whether additional controls are necessary for RISC-2 SSCS

5.6.3 Evaluating whether additional controls are necessary for RISC-1 SSCs to ensure acceptable performance for beyond design basis functions.

5.6.4 Implementing alternative treatment requirements or other changes as identified above.

5.7 The Manager, Nuclear Licensing is responsible for ensuring that the following requirements in 10 CFR 50.69 are adhered to:

5.7.1 Following the implementation of 10 CFR 50.69, the Final Safety Analysis Report shall be updated to reflect which systems have been categorized (from 10 CFR 50.69, part f.2)

5.7.2 Submitting 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 (from 10 CFR 50.69, part g).

6.0 Procedure

6.1 This procedure provides a summary of categorization process and a summary of application of alternative treatment requirements that can be implemented after final risk categories are applied to each component in a system. The Nuclear Licensing (NL) department will update the Final Safety Analysis Report when treatments are implemented. The NL department will also 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.

6.2 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).

Instructions NMP-ES-065-001 (10CFR50.69 Active Component Risk Significance Insights), NMP-ES-065-002 (Passive Component Categorization), and NMP-ES-065-003 (Risk Informed Categorization for Systems, Structures, and Components) are associated with NMP-ES-065. These instructions determine safety significance (High Safety Significant or Low Safety Significant) of each component for a selected system using methods identified in these instructions. The preliminary results will determine the risk categories (e.g., RISC-1, RISC-2, RISC-3, and RISC-4) for each component in a system.

These results are sent to the Integrated Decision Making Panel (NMP-ES-066 and NMP-ES-066-001). The panel will review and approve the results.

Attachment A shows the above relationship.

6.3 Requirements

The following are the requirements that MUST be met before categorization of a system is performed.

6.3.1 Training

Specific training and qualifications requirements for IDP members and designated alternates is detailed in NMP-ES-066-001.

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.

6.3.2 PRA Capability

The risk-informed categorization of SSCs in nuclear power plant applications requires the use of an appropriately detailed PRA of sound technical quality. At a minimum, the PRA must model severe accident scenarios resulting from internal initiating events occurring at full power operation. Importance measures related to core damage frequency (CDF) and large early release frequency (LERF) are used to identify safety significant SSCs. In addition, other risk contributors must also be 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. Sensitivity studies are performed for LSS PRA-modeled components to ensure sufficient margins exist.

6.4 Summary of Categorization Process

6.4.1 Risk Categories

SSCs shall be categorized as RISC-1, RISC-2, RISC-3, or RISC-4.

6.4.2 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).

6.4.3 Qualitative Insights

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

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

6.4.5 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.

6.4.6 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.

6.4.7 Risk Significant Attributes

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

6.4.8 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).

6.4.9 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.

NOTE

1. **Refer to NMP-ES-065-004 for detailed information on guidance related to Treatment (instructions to be developed).**
2. **Following the implementation of 10 CR 50.69, the Nuclear Licensing Department shall update the Final Safety Analysis Report to reflect which systems have been categorized.**
3. **The Nuclear Licensing Department shall submitting 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**

6.5 Summary of Application of Alternative Treatment Requirements

6.5.1 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]

6.5.2 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."

6.5.3 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.

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

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

- Identify existing special treatment program scope and purpose
- Identify sources of existing program and treatment requirements
- Identify requirements that no longer apply per 10 CFR 50.69
- Identify alternative treatment elements that support the design basis
- Develop and implement alternative treatment options

6.5.6 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.

6.5.7 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 <u>beyond design basis functions</u>.

6.5.8 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

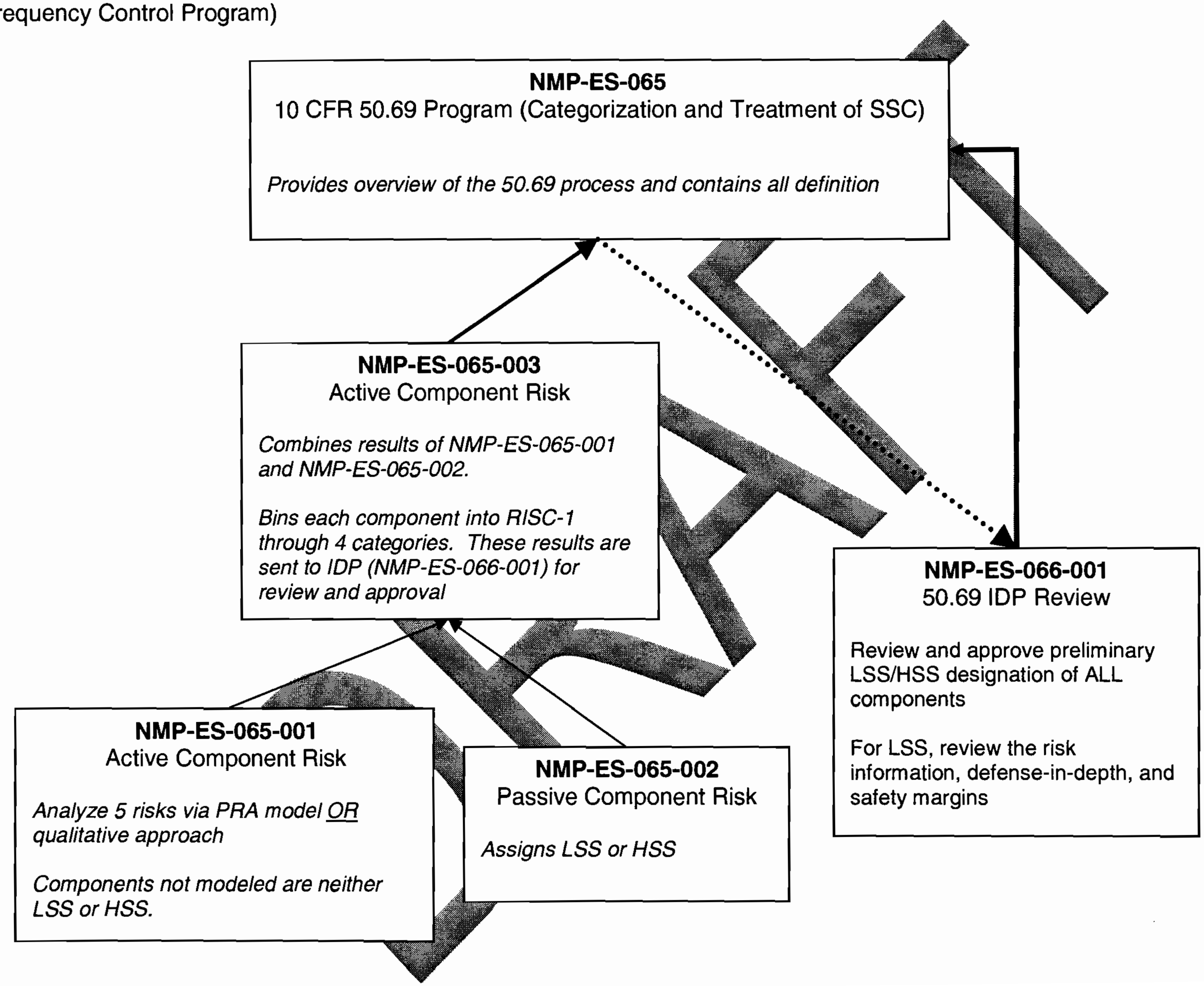
7.0 **Records**

This procedure itself does not generate records. However, instructions associated with this procedure generate records. These instructions outline how records will be maintained.

8.0 **Commitments**

None

_**Attachment 1**: Summary of relationship of this procedure (NMP-ES-065) with associated instructions <u>and</u> 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)



Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 2

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

Instruction Owner: ____
(Print: Name / Title / Site)

Approved By: ____
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates: ________ ________ ________ ________ ________
Corporate FNP HNP VEGP 1-2 VEGP 3-4

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

Writer(s):

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

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on site for reference as needed.	**ALL**

Revision Description

Version Number	Revision Description
1.0	Initial issue

1.0 Purpose

The purpose of this 10CFR50.69 Active Component Risk Significance Instruction is to promote effective, consistent use of the 10CFR50.69 program across the SNC fleet.

This instruction includes requirements and instructions for 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*.

This instruction is part of an integrated categorization process which includes the following 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 Risk Insights
- NMP-ES-065-003, 10 CFR 50.69 Risk Significance Categorization for Systems, Structures, and Components
- 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
- NMP-ES-066-001, Integrated Decision-Making Panel For Risk Informed SSC Categorization: Duties And Responsibilities

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) and which are discussed separately in instruction NMP-ES-065-004.

NOTE: This instruction has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Categorization activities described in this instruction may be performed prior to NRC approval of the license amendment. However, the alternative treatment requirements specified in 10 CFR 50.69 (d) shall NOT be implemented UNLESS the following actions are verified to be completed:

After the license amendment is approved by the NRC, an evaluation shall be performed and documented to ensure that the process described in this instruction meets the requirements of, and is consistent with, the NRC-approved license amendment. The performance of this evaluation shall be tracked via a Condition Report action. This evaluation shall be approved by the Manager, Risk-Informed Engineering and by the Manager, Licensing. The instruction shall then be revised at this time to remove this Section.

IF the above evaluation concludes that the process described in this instruction does not meet the requirements of, or is inconsistent with, the approved license amendment, THEN this instruction shall be revised accordingly and any evaluations or activities already performed shall be re-performed using the revised procedural requirements.

2.0 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.

This instruction was created and is maintained under the direction of the Risk-Informed Engineering Manager.

3.0 References

3.1 10 CFR 50.69, "Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors"

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

3.3 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

3.4 NMP-ES-065, 10 CFR 50.69 Program

3.5 NMP-ES-065-002, Passive Risk Insights

3.6 NMP-ES-065-003, 10CFR50.69 Risk Informed Categorization for Systems, Structures, and Components

3.7 NMP-ES-065-004, Alternative Treatment Requirements

3.8 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

3.9 NMP-ES-066-001, Integrated Decision-Making Panel For Risk Informed SSC Categorization: Duties And Responsibilities

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

3.11 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

3.12 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.

4.0 Definitions

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

5.0 Responsibilities

5.1 Responsibilities for the 10CFR50.69 Process are found in NMP-ES-065.

5.2 The cognizant Risk-Informed Application engineer is responsible for the following activities associated with the active SSC risk significance process.

5.2.1 Providing the internal events at power PRA base case risk importances for SSCs in the system under review, for system SSCs modeled in the PRA and system SSCs not modeled in the PRA.

5.2.2 Providing 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.

5.2.3 Providing the results of the integrated risk importance analysis for SSCs in the system under review.

5.2.4 Providing 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.

5.2.5 Providing additional PRA Model insights which may influence the SSC categorization outcome.

5.2.6 Providing PRA risk changes, resulting from model updates or other factors that could impact existing SSC categorizations.

5.2.7 Over time, participating in the periodic performance review process and analyzing the impact of changes in performance of SSCs categorized as LSS on the risk significance results.

5.3 The cognizant System Engineer is responsible for the following activities associated with the active SSC risk significance process.

5.3.1 Providing the list of systems, functions, and associated SSCs for which risk significance information is required.

5.3.2 Providing design basis and severe accident functions of SSCs relative to each hazard evaluated.

6.0 Procedure

6.1 Requirements

6.1.1 Risk Categories

SSCs shall be categorized as HSS or LSS using the categorization process outlined in this instruction.

6.1.2 PRA Capability

The plant internal events at power PRA model of record is used in this assessment.

The risk-informed categorization of SSCs in nuclear power plant applications requires the use of an appropriately detailed PRA of sound technical quality. At a minimum, the PRA must model severe accident scenarios resulting from internal initiating events occurring at full power operation. NRC expectations for PRA capability for 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 (e.g., RA-Sa-2009 -- Ref. 3.12 -- or subsequent revisions) as endorsed with NRC clarifications in Reg Guide 1.200 (Ref. 3.11), and shown to meet most requirements in that standard at capability category II or better. If there are areas where the PRA does not meet a requirement at capability category II, an assessment should be made, and documented, regarding the potential impact of such limitations on the 50.69 categorization application and the manner in which they will be compensated for in using the PRA. A similar confirmation of technical adequacy is required for each PRA model used in the categorization process (e.g., internal events at power, internal fire, seismic, etc.).

In using the PRA for 50.69 categorization, a characterization of the adequacy of the PRA, as well as PRA limitations, must be stated as part of the presentation of categorization results to the IDP as a basis for the adequacy of the risk information used in the categorization process. Such limitations might include hazards that are not modeled (e.g., external initiating events), plant shutdown risks, and SSCs that are not modeled.

6.1.3 Determination of SSC Importances

The assessment of importance for an SSC involves the identification of PRA basic events that represent the SSC. This can include:

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

The PRA analyst must identify the events in the PRA that can be used to represent each SSC. Within this mapping, record whether the PRA explicitly models the performance of the SSC (e.g., pump X fails to start), implicitly models SSC (e.g., via assumption for availability to support a human action, as a contributor to an initiating event, etc.) or a combination of both types of events.

The contribution of common cause to a component's importance must also be addressed. If a component does not have a common cause basic event in the PRA to be included in the computation of importances, then an assessment should be made as to whether a common cause event should be added to the model.

6.1.4 Availability of PRA models for Risk Contributor

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.

6.1.4.1 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.

6.1.4.2 **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 Steps 6.3 and 6.4.

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

6.1.4.4 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 is lowered.

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.

6.2 Risk Characterization Overview [per NEI-00-04, Ref. 3.2]

The NEI 00-04 categorization process addresses a full scope of hazards, as well as plant shutdown safety. Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is initially assessed separately from each of five risk perspectives, and then an integrated risk significance evaluation is used to identify SSCs that are potentially safety significant for consideration by the IDP. The 5 risk perspectives are:

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

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

Table 6-1 provides a summary of the alternative approaches taken to address each risk contributor. A brief description of each of these aspects is described in the following paragraphs.

Table 6-1
Summary of Risk Significance Characterization Used in NEI 00-04

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

6.3 Risk Evaluations based on PRA or Other Hazards Analyses

6.3.1 The process for assessing risk hazards identified in Table 6-1 is defined below (sections 6.3.2 through 6.3.14), consistent with NEI 00-04 (Ref. 3.2). This process will provide the following risk assessment results to be provided as input to the overall categorization of SSCs.

- For components that are modeled by one or more PRAs, an integrated importance assessment (per 6.3.14) 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).

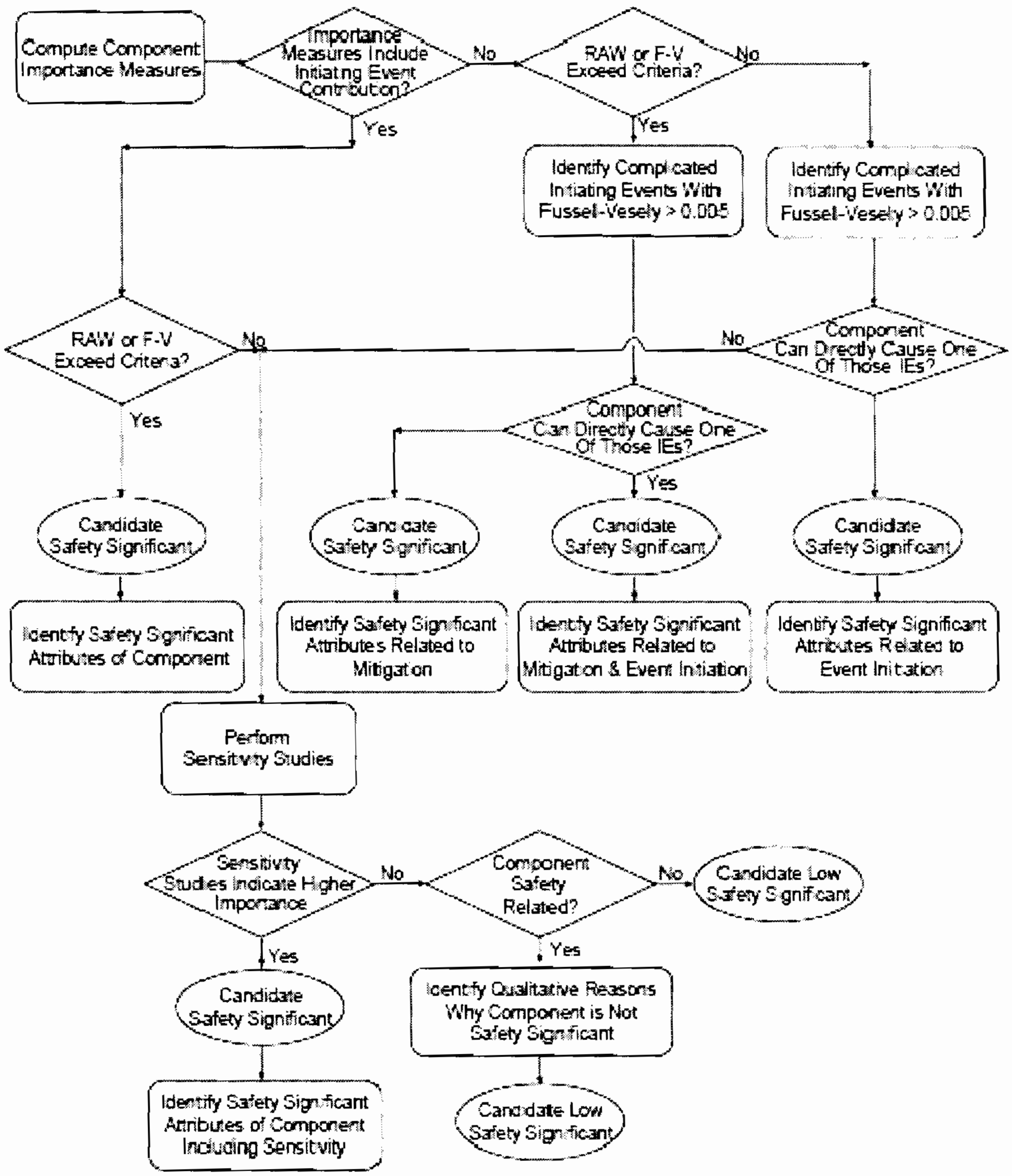
6.3.2 Internal Events at Power Risk Importance Using the Internal Events at Power PRA

The use of the internal events at power PRA to quantify the risk importance measures for the identified functions and SSCs in the system of interest is described in this section. The overall process is shown in Figure 6-1, per NEI-00-04. This risk importance process, including sensitivity studies, is performed for both CDF and LERF. Components being categorized must satisfy the risk importance criteria described in Table 6-2 for both CDF and LERF in order to be candidate LSS.

Table 6-2 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

Figure 6-1 (NEI-00-04 Figure 5-2)

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



NOTES

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.

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.

6.3.2.1 Identify the PRA basic events that represent the SSCs of interest.

6.3.2.2 Create a mapping of those components to be categorized to the events in the PRA that can be used to represent each component.

a) 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.
b) If a component of interest does not have a common cause event in the PRA to be included in the computation of importances, then an assessment should be made as to whether a common cause event should be added to the model.

6.3.2.3 Determine if 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?

a) If so, the PRA importance measures provide sufficient scope to perform the initial screening. Steps 6.3.2.6 through 6.3.2.8 define the component's candidate safety significance.
b) If not, additional evaluation as defined in Step 6.3.2.9 is required.

6.3.2.4 **If the PRA model importance accounts for the contribution in initiating events**, then for each component of interest, use the internal events at power PRA to calculate the F-V and RAW for that component.

a) 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.
   - 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.

b) The RAW importance of a component is the maximum of the RAW values computed for basic events involving failure modes of the individual component.

c) 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.

6.3.2.5 **If the PRA model importance accounts for the contribution in initiating events** and if *any* of the risk importance criteria listed in Table 6-2 are exceeded for a component, that component is considered *candidate high safety-significant*, and its safety significant attributes must be documented. Table 6-3 provides examples of the use of these criteria.

6.3.2.6 **If the PRA model importance accounts for the contribution in initiating events** and if the component's risk importances are less than each of the criteria in Table 6-2, then include the component in the set of *potential* candidate LSS components for which sensitivity studies are to be performed (Step 6.3.3).

6.3.2.7 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.

a) Determine whether the component exceeds any of the risk importance criteria in Table 6-2.

b) **If so**, the component is *candidate safety-significant*. 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.
   1. 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.
   2. If the component *cannot* directly cause a complicated initiating event with F-V > 0.005, then document the component's safety significant attributes relative to mitigation.

c) **If not**, then:
   1. If the component *can* directly cause a complicated initiating event with F-V > 0.005, then the component is candidate safety significant. Document the component's safety significant attributes relative to event initiation.
   2. If the component *cannot* directly cause a complicated initiating event with F-V > 0.005, then include the component in the set of *potential* candidate LSS components for which sensitivity studies are to be performed (Step 6.3.3).

Table 6-3 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.			

6.3.3 **Internal Events at Power PRA Sensitivity Studies**

6.3.3.1 If the importance measures computed by the PRA tool indicate that ALL components, including non-safety-related components, are HSS, then the recommended sensitivity studies are not needed for the system that is being categorized.

However, if the importance measures computed by the PRA tool do not indicate that a component meets the F-V or RAW criteria for HSS (i.e., may be candidate 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.

If an SSC that had been initially identified as candidate LSS is found to exceed the safety significance thresholds in one of the specified sensitivity studies, this information is to be documented as part of the information package to be considered in the risk significance categorization (per Ref. 3.6). This information package is ultimately provided to the IDP (per Ref. 3.8) for consideration, along with an explanation of the results of the sensitivity study.

6.3.3.2 The recommended sensitivity studies for internal events PRA are identified in Table 6-4.

a) The sensitivity studies on human error rates, common cause failures, and maintenance unavailabilities are performed to ensure that assumptions of the PRA are not masking the importance of an SSC. In these sensitivities, the indicated changes are made 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 95th percentile values are used for ALL HEPs in the PRA.

b) In cases where plant-specific uncertainty distributions are not readily available, other PRAs should be reviewed to identify appropriate parameter ranges. Experience with plant-specific PRAs has shown that the variations in distributions are relatively small, especially with respect the ratio of the mean and 95th percentile values in lognormal distributions (the most common distribution used in PRAs). 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.

c) If the sensitivity studies identify that the component could be safety-significant, then the safety-significant attributes that yielded that conclusion should be identified.

Table 6-4 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.

6.3.3.3 If, following the sensitivity studies, the component is still found to be LSS from an internal events perspective, it is a candidate for RISC-3 or RISC-4. In this case the analyst is to identify 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.). The component is retained as candidate low safety significant from an internal events at power risk perspective.

6.3.4 **Internal Fire Risk Importance Evaluation using Fire PRA**

6.3.4.1 For plants with a fire PRA, the generalized safety significance process is the same as the process for an internal events at power PRA. This process is shown on Figure 6-2, and is discussed in the following steps.

> **NOTE**
>
> 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.

6.3.4.2 Use the Fire PRA to quantify the fire risk importance measures for the identified SSCs in the system of interest. The overall process is shown in Figure 6-2, per NEI-00-04.

> **NOTE**
>
> If the fire PRA CDF, including all screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), then safety significance of SSCs considered in the fire PRA can be considered LSS from a fire perspective.

> **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.

6.3.4.3 This risk importance process is performed for 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.

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

6.3.4.5 If any of the risk importance criteria listed in Table 6-2 are exceeded for a component, that component is considered *candidate high safety-significant*, and its safety significant attributes must be documented. Table 6-3 provides additional guidance in evaluating risk importance results.

6.3.5 Internal Fire PRA Risk Importance Sensitivity Studies

6.3.5.1 If the component's risk importances are less than each of the criteria in Table 6-3, then perform the recommended fire PRA sensitivity studies (as identified in Table 6-5).

6.3.5.2 If the sensitivity studies identify that the component could be safety-significant, then the component is designated as *candidate high safety-significant from a fire risk perspective* and the attributes which yielded that conclusion should be identified.

6.3.5.3 If the sensitivity studies confirm that the component's risk importances are less than each of the criteria in Table 6-2, then the component *may be candidate low safety significant from a fire risk perspective*.

a) If such a component is not safety related, then it is *candidate low safety significant from a fire risk perspective*.

b) If such a component is safety-related, then qualitative reasons must be identified 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 the component is retained as *candidate low safety significant from a fire risk perspective*.

Table 6-5 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.

6.3.6 Internal Fire Safety Significance Without Fire PRA

6.3.6.1 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.

NOTE

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.

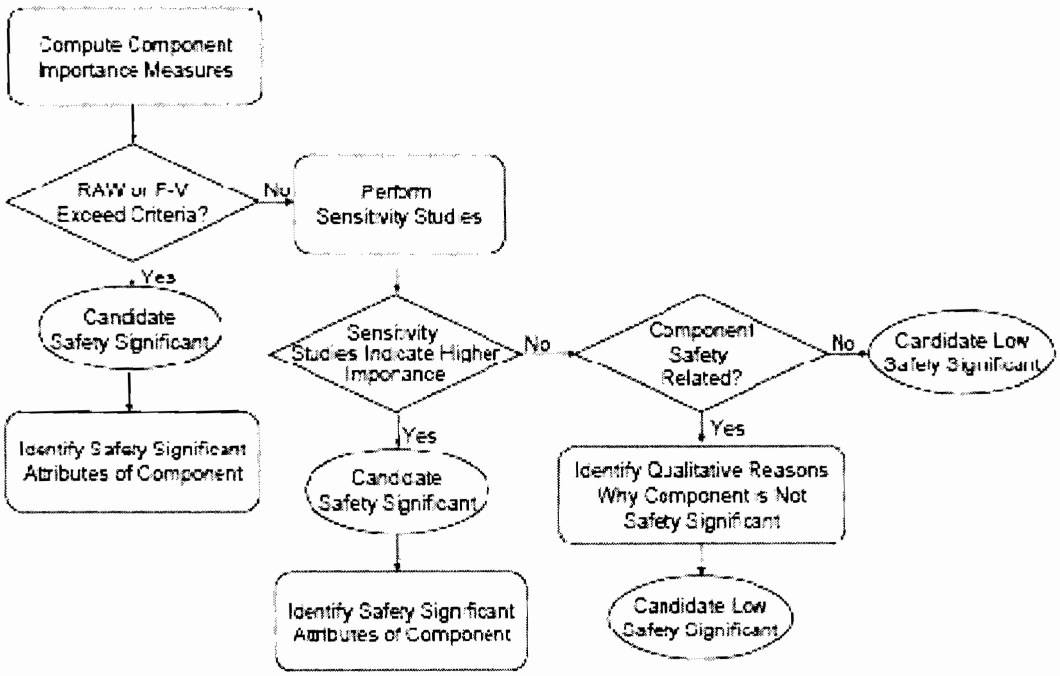
6.3.6.2 For each component, identify the fire design basis and severe accident functions of the component.

6.3.6.3 Review the plant's Fire Safe Shutdown analysis to determine if the component is credited as part of the safe shutdown paths evaluated.

a) If a component is credited as part of a fire safe shutdown path, it is considered *safety-significant from a fire risk perspective*, and the attributes which yielded that conclusion should be identified. 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.
b) If the component does not participate in the safe shutdown path, then it is considered a *candidate low safety-significant with respect to internal fire risk.*

Figure 6-2

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



6.3.7 **Seismic Risk Importance Evaluation using Seismic PRA**

6.3.7.1 For plants with a seismic PRA, the generalized safety significance process is the same as the process for an internal events at power PRA. This process is shown on Figure 6-2, and is discussed in the following steps.

NOTE

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.

6.3.7.2 Use the seismic PRA to quantify the fire risk importance measures for the identified SSCs in the system of interest. The overall process is shown in Figure 6-2, per NEI-00-04.

NOTE

If the seismic PRA CDF, including all screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), then safety significance of SSCs considered in the seismic PRA can be considered LSS from a seismic perspective.

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 RISC-3 or RISC-4 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.

6.3.7.3 This risk importance process is performed for 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.

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

6.3.7.5 If any of the risk importance criteria in Table 6-2 are exceeded for a component, that component is considered *candidate high safety-significant*, and its safety significant attributes must be documented. Table 6-3 provides additional guidance in evaluating risk importance results.

6.3.8 **Seismic PRA Risk Importance Sensitivity Studies**

6.3.8.1 If the component's risk importances are less than each of the criteria in Table 6-2, then perform the recommended seismic PRA sensitivity studies (as identified in Table 6-6).

6.3.8.2 If the sensitivity studies identify that the component could be safety-significant, then the component is designated as *candidate high safety-significant from a seismic risk perspective* and the attributes which yielded that conclusion should be identified.

6.3.8.3 If the sensitivity studies confirm that the component's risk importances are less than each of the criteria in Table 6-2, then the component *may be candidate low safety significant from a seismic risk perspective.*

a) If such a component is not safety related, then it is *candidate low safety significant from a seismic risk perspective*.

b) If such a component is safety-related, then qualitative reasons must be identified 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 the component is retained as *candidate low safety significant from a seismic risk perspective*.

Table 6-6 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.

6.3.9 Seismic Safety Significance Without Seismic PRA

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.

6.3.9.1 For each component, identify the seismic design basis and severe accident functions of the component.

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

a) If a component is credited as part of a seismic-margins-evaluated safe shutdown path, it is considered *safety-significant from a seismic risk perspective*, and the attributes which yielded that conclusion should be identified.

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

6.3.10 Other External Hazards Risk Evaluation Using PRA

6.3.10.1 For plants with a PRA that evaluates other external hazards, the generalized safety significance process is as shown on Figure 6-2, and is discussed in the following steps.

6.3.10.2 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.

6.3.10.3 If the system or structure is determined to be evaluated in the external hazards PRA, then the following steps are used to determine candidate safety significance.

NOTE

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.

6.3.10.4 Use the other external hazards PRA to quantify the external hazards risk importance measures for the identified SSCs in the system of interest.

> **NOTE**
> If the other external hazards PRA CDF, including all screened scenarios, is a small fraction of the internal events at power CDF (i.e., <1%), then safety significance of SSCs considered in the external hazards PRA can be considered LSS from an other external hazard perspective.

6.3.10.5 This risk importance process is performed for 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.

6.3.10.6 Follow the evaluation process steps for seismic risk importance evaluation, in section 6.3.7 and 6.3.8 (sensitivity studies as indicated in Table 6-6; note that the sensitivity for "correlated fragilities" applies and should be interpreted as fragilities related to the other hazard in question).

6.3.11 Other External Hazards Risk Evaluation Without PRA

6.3.11.1 If the plant does not have an external hazards PRA, then use 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.

6.3.11.2 If the SSC is evaluated in the external hazards screening analysis, then the following steps are used to determine candidate safety significance.

6.3.11.3 For each component, identify the other external hazard design basis and severe accident functions of the component.

6.3.11.4 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.

a) If a component is credited as part of an other external hazards-evaluated safe shutdown path, it is considered *safety-significant from an other external hazards perspective*, and the attributes which yielded that conclusion should be identified.
b) If the component does not participate in an other external hazards evaluated shutdown path, it is *candidate low safety-significant with respect to the other external hazard risk, IF* it can be shown that:

- 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

> **NOTE**
> If a system/structure is not involved in either an external hazards PRA or external hazards screening evaluation, then the SSC is categorized as *candidate LSS from the standpoint of other external risks.*

6.3.12 **Shutdown Safety Assessment Using Shutdown PRA**

6.3.12.1 For plants with a shutdown PRA that is comparable to an at-power PRA (i.e., generates annual average CDF/LERF), the generalized safety significance process is the same as the process for an internal events at power PRA. This process is shown in Figure 6-1.

6.3.12.2 Follow the process defined in steps 6.3.2 and 6.3.3 using the shutdown PRA.

> **NOTE**
> If the shutdown PRA CDF is a small fraction of the internal events at power CDF (i.e., <1%), then safety significance of SSCs considered in the shutdown PRA can be considered LSS from a shutdown perspective.

6.3.13 **Shutdown Safety Assessment Using NUMARC 91-06 Program**

6.3.13.1 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. In the shutdown safety assessment process guidance provided in NEI-00-04, a component is identified as *safety-significant for shutdown conditions* for either of the following reasons:

a) When multiple systems/trains are available to satisfy the key safety function, those SSCs that support the primary and first alternative methods to satisfy the key safety function are considered to be the "primary shutdown safety system" and are thus *candidate safety-significant with respect to shutdown risk*.

> **NOTE**
>
> 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.

b) If the component's failure would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.), it is *candidate safety-significant with respect to shutdown risk*.

6.3.13.2 If the component does not participate in either of the manners identified in 6.3.13.1, then it is considered *candidate low safety significance with respect to shutdown safety*.

6.3.14 **Integral Assessment of Overall Risk Significance**

6.3.14.1 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 all of the PRA-modeled SSCs, not only those that have already been determined to be safety significant. However, the integrated importance cannot be higher than the maximum of the individual measures.

6.3.14.2 The integrated importance measure weights the importance from each risk contributor (e.g., internal events, fire, seismic PRAs) by the fraction of the total core damage frequency contributed by that contributor. The following formulas define how such measures are to be computed for CDF.

<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

6.3.14.3 Once calculated, an assessment should be made of these integrated values against the screening criteria of F-V >0.005, RAW > 2.0 for individual basic events, and RAW > 20 for common cause basic events.

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

6.3.14.4 The same process should be used for LERF, if available.

6.3.14.5 The results of the integrated assessment should be documented and reported 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.

6.4 Risk Sensitivity Study

6.4.1 The following provides background information regarding risk sensitivity studies for evaluation of the risk implications of changes in special treatment.

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.

6.4.1.1 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.

- 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.

NOTE

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

6.4.1.2 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.

6.4.2 **Perform Initial Sensitivity Study**

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

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

6.4.2.3 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.

6.4.2.4 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, it may also be advisable to perform 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.

- 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.

6.4.2.5 Determine if the recommended FV and RAW threshold values used in the screening need to be changed based on results of this sensitivity study.

- If the risk evaluation shows that the changes in CDF and LERF as a result of changes in special treatment requirements are not within the acceptance guidelines of the Regulatory Guide 1.174, then a lower F-V threshold value may be needed (e.g., > 0.0025 = 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.

6.4.3 **Perform cumulative sensitivity for all the SSCs categorized using this process.**

6.4.3.1 Repeat the above process to evaluate the cumulative impact of all LSS components for all systems that have been categorized.

6.4.4 Provide results of individual system and cumulative sensitivity studies to the IDP

6.4.4.1 This should provide the IDP with both the overall assessment of the potential risk implications and the relative contribution of each system.

6.4.5 **Re-evaluate sensitivities after IDP consideration**

6.4.5.1 The sensitivity studies should be checked and revised when the IDP has completed its final categorization if the IDP has changed SSC categorizations, to assure that the conclusions regarding the potential aggregate impact have not changed significantly.

6.4.5.2 If the categorization of SSCs is done at different times, the sensitivity study should consider the potential cumulative impact of all SSCs categorized, not individual systems or components.

NOTE

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

6.5 Reviews and Performance Feedback

6.5.1 **Perform PRA Reviews to ensure continued validity of categorization results and to review SSC performance.**

6.5.1.1 Periodic update of the PRA (at least once per every other refueling outage for Unit 1) must be performed, after which a review must be done for 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.

Additional reviews, in addition to the periodic reviews, may be needed if a PRA model or other risk information is upgraded (as defined in Ref. 3.12). In such cases, a post-model-upgrade review of the SSC categorization should also be performed to determine if previously-performed categorization results may be affected by the model changes.

6.5.1.2 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) 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.

(b) 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 chance in categorization would not be indicated.

When a change to the categorization of an SSC is suggested by a change in the PRA model as determined from the absolute importance measures, such changes should be presented to the IDP for concurrence.

7.0 Records

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

8.0 Commitments

None

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 3

Draft NMP-ES-065-002
10 CFR 50.69 Passive Component Categorization

Instruction Owner: ______________________________
(Print: Name / Title / Site)

Approved By: ______________________________
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:

______	______	______	______	______
Corporate	FNP	HNP	VEGP 1-2	VEGP 3-4

PRB review is required for this instruction

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on site for reference as needed.	**ALL**

Instruction Version Description

Version Number	Version Description
1.0	Initial Issue

Table of Contents

Page

1.0 Purpose

The purpose of this 10CFR50.69 Passive Component Categorization Instruction is to promote effective, consistent use of the 10CFR50.69 program across the SNC fleet.

This instruction includes requirements and instructions for the development and review of the risk-informed categorization of Passive Components in support of a 10CFR50.69 application.

2.0 Applicability

This instruction is applicable only to those plant systems that have been selected for categorization and contain passive components.

This instruction is applicable to activities involving the 10CFR50.69 Passive Component Categorization performed by Southern Nuclear Operating Company (SNC) personnel or supplemental personnel.

3.0 References

3.1 NMP-ES-065, 10 CFR 50.69 Program

3.2 NMP-ES-065-001, Active 10CFR50.69 Active Component Risk Significance Insights

3.3 NMP-ES-065-003, 10CFR50.69 Risk Informed Categorization for Systems, Structures, and Components

3.4 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.

3.5 NMP-ES-066-001, Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

3.6 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.

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

3.8 10CFR50.69 Final Rule, *Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors*, November 22, 2004.

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

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

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

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

4.0 Definitions

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

5.0 Responsibilities

Responsibilities for the 10CFR50.69 Process are found in NMP-ES-065.

6.0 Procedure

> **Note:**
>
> The source documents for methodology mentioned in this instruction is EPRI Report TR-112657, Rev B-A.
>
> IF further details on the evaluation of operator actions and its impact on the consequence ranking; the evaluation and ranking of the consequence impact groups; and configurations and the evaluation of shutdown and external events are needed, 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, contact Risk-Informed Engineering Department.

6.1 General Requirements

6.1.1 Scope

6.1.1.1 The process for determining the Passive Component Categorization shall be applied on a system basis, including all components and their associated supports within the selected system(s).

6.1.1.2 This process is applied to Class 2, 3 and non Class systems or their associated supports (exclusive of Class CC and MC items).

6.1.2 Attachment A provides an overview of the Passive Component Categorization process

6.1.3 Categorization
Components and component supports in systems subject to the evaluation contained in this instruction shall be classified High Safety Significant (HSS) or Low Safety Significant (LSS) in accordance with sections 6.2 and 6.3.

6.1.4 Required Disciplines

Necessary personnel to perform the risk-informed safety classification evaluation, review and documentation should be made available. Personnel with expertise in the following disciplines should be included:

(a) probabilistic risk assessment (PRA)

(b) plant operations

(c) system design

(d) safety or accident analysis

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

6.2 Consequence Evaluation

6.2.1 General Information

6.2.1.1 To ease the analysis and documentation burden, components may be grouped 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 shall be assessed for each piping segment.

6.2.1.2 A Consequence Category for each piping segment is determined from the Failure Modes and Effects Analysis (FMEA) and Impact Group Assessment as defined in sections 6.2.2 and 6.2.3, respectively.

6.2.1.3 Throughout the evaluations of sections 6.2 and 6.3, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the segment under consideration. To take credit for operator actions, the following features shall be provided:

6.2.1.3.1 An alarm or other system feature to provide clear indication of failure,

6.2.1.3.2 Equipment activated to recover from the condition must not be affected by the failure,

6.2.1.3.3 Time duration and resources are sufficient to perform operator action,

6.2.1.3.4 Plant procedures to define operator actions, and

6.2.1.3.5 Operator training in the procedures.

6.2.1.4 Success criteria diagrams shall be developed for all relevant initiating events.

6.2.2 Failure Modes and Effects Analysis (FMEA)

Identify potential failure modes for each system **OR** piping segment and evaluate their effects. This evaluation shall consider the following:

6.2.2.1 Pressure Boundary Failure Size - The consequence evaluation shall 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 shall 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).

6.2.2.2 Isolability of the Break - 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 6.2.1.3. highest consequence ranking shall be used.

6.2.2.3 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).

6.2.2.4 Initiating Events - I 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.

6.2.2.5 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 shall be evaluated.

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

6.2.2.7 System Configuration - The consequence evaluation and ranking is organized into four basic consequence impact groups as discussed in section 6.2.3. The three corresponding system configurations for these impact groups are defined in Table 6.

6.2.3 Impact Group Assessment

The results of the FMEA evaluation for each system, or portion thereof, shall be classified into one of three core damage Impact Groups: initiating event, system, or combination. In addition, failures shall also be evaluated for their importance relative to containment performance.

Each system, or portion thereof, shall be partitioned into postulated piping failures that cause an initiating event, disable a system/train/loop without causing an initiating event, or cause an initiating event and disable a system/train/loop.

Evaluations in steps 6.2.3.1 through 6.2.3.3 determine the failure's importance relative to core damage.

The consequence category assignment (high, medium, low, or none) for each piping segment within each impact group shall be determined in accordance with the following.

6.2.3.1 Initiating Event (IE) Impact Group Assessment

When the postulated failure results in only an initiating event (e.g., loss of feedwater, reactor trip), the consequence shall be classified into one of four categories: high, medium, low, or none. The initiating event category shall be assigned 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.

6.2.3.2 System Impact Group Assessment

The consequence category of a failure that does not cause an initiating event, but degrades or fails a system/train/loop essential to prevention of core damage, shall be evaluated. This evaluation shall include all safety functions supported by the segment as well as all safety functions impacted by the failure of the segment. This evaluation shall be 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, which determines 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.

Consequence categories shall be assigned in accordance with Table 2 as High, Medium, or Low. 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. Additionally, for defense in depth purposes, all postulated failures leading to "zero defense" (i.e., no backup trains) shall be assigned a high consequence.

6.2.3.3 Combination Impact Group Assessment

The consequence category for a piping segment whose failure results in both an initiating event and the degradation or loss of a system shall be determined using Table 3. The consequence category is a function of two factors:

- Use of the system to mitigate the induced initiating event;
- 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.

6.2.3.4 Containment Performance Impact Group Assessment
The previously established consequence rank (6.2.3.1, 6.2.3.2, or 6.2.3.3) shall be reviewed and adjusted to reflect the pressure boundary failure's impact on in containment performance. This impact shall be evaluated as follows:

- Table 4 shall be used to assign consequence categories for those piping failures that can lead to a LOCA that bypasses containment.
- For postulated failures that do not result in a LOCA that bypasses containment, the quantitative indices of Table 5 for CLERP shall be used.

6.2.3.5 Shutdown operation shall be evaluated. The previously established consequence rank shall be reviewed and adjusted to reflect the pressure boundary failure's impact on plant operation during shutdown.

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

If a shutdown PRA is not available, the effect of pressure-boundary failures on core damage and containment performance shall be evaluated.

The major characteristics to be considered are defined as follows:

- 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.

6.2.3.6 External events shall be evaluated. The previously established consequence rank shall be reviewed and adjusted to reflect the pressure boundary failure's impact on the mitigation of external events. The effect of external events on core damage and containment performance shall be evaluated 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).

6.3 Classification

Piping segments may be grouped together within a system, if the analysis and assessment performed in section 6.2 determines the effect of the postulated failures to be the same. The classification shall be as follows:

<u>Classification Definitions</u>

HSS – Piping segment considered high-safety-significant

LSS – Piping segment considered low-safety-significant

6.3.1 Classification Considerations

6.3.1.1 Piping segments determined to be a High consequence category in any table by the analysis and assessment in section 6.2 shall be considered **HSS**.

6.3.1.2 Piping segments determined to be a Medium, Low, or None (no change to base case) consequence category in any table by the consequence evaluation in section 6.2 shall be determined to be HSS or LSS by considering the information in 6.3.1.2.1 through 6.3.1.2.6 below. Under the same conditions of section 6.2.2.1, a large pressure boundary leak does not need to be assumed. Also, credit may be taken for plant features and operator actions to the extent these would not be affected by failure of the segment under consideration. If plant features and operator actions are credited, they shall be consistent with those credited in section 6.2.1.3.

The following conditions shall be evaluated and answered TRUE or FALSE.

6.3.1.2.1 Failure of the pressure retaining function of the segment will not directly or indirectly (e.g., through spatial effects) fail a basic safety function.

6.3.1.2.2 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.

6.3.1.2.3 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.

6.3.1.2.4 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.

6.3.1.2.5 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.

6.3.1.2.6 This process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth is maintained if:

- 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** HSS shall be assigned. Otherwise, LSS shall be assigned.

6.3.1.3 If LSS has been assigned from section 6.3.1.2, then this instruction shall verify that there are sufficient margins to account for uncertainty in the engineering analysis and in the supporting data. 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. If sufficient margins are maintained then LSS should be assigned; if not, then HSS shall be assigned.

6.3.1.4 A component support, hanger, or snubber shall have the same classification as the highest-ranked piping segment within the piping stress analytical model in which the support is included.

7.0 Records

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

8.0 Commitments

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 6.2.3.1

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
$10^{-6} < \text{value} \leq 10^{-4}$	$10^{-7} < \text{value} \leq 10^{-5}$	Medium
$\leq 10^{-6}$	$\leq 10^{-7}$	Low
No change to base case	No change to base case	None

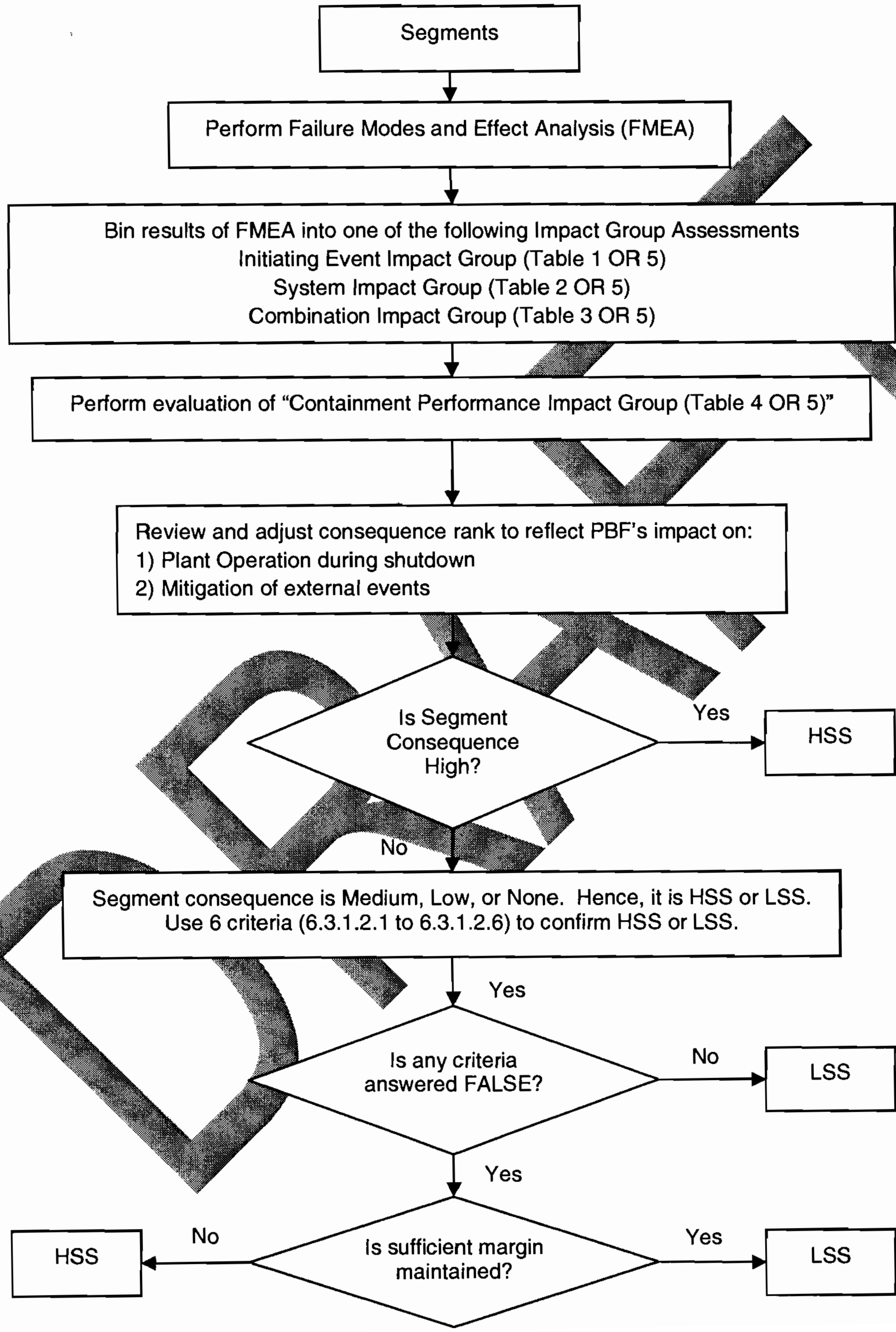
Table 6
Definition of Consequence Impact Groups and Configurations

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 not 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
	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



Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 4

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

Instruction Owner: ______________________________
(Print: Name / Title / Site)

Approved By: ______________________________
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates: ________ Corporate ________ FNP ________ HNP ________ VEGP 1-2 ________ VEGP 3-4

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

Writer(s):

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

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on site for reference as needed.	**ALL**

Revision Description

Version Number	Revision Description
1.0	Initial issue

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1.0 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, 10CFR50.69 Active Component Risk Significance Insights
- NMP-ES-065-002, 10CFR50.69 Passive Component Categorization
- NMP-ES-066-001, 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) and which are discussed separately in instruction NMP-ES-065-004.

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

1.5 This instruction has been developed in anticipation of NRC approval of a license amendment request to adopt 10 CFR 50.69. Categorization activities described in this instruction may be performed prior to NRC approval of the license amendment. However, the alternative treatment requirements specified in 10 CFR 50.69 (d) shall **NOT** be implemented **UNLESS** the following actions are verified to be completed:

1.5.1 After the license amendment is approved by the NRC, an evaluation shall be performed and documented to ensure that the process described in this instruction meets the requirements of, and is consistent with, the NRC-approved license amendment. The performance of this evaluation shall be tracked via a Condition Report action. This evaluation shall be approved by the Manager, Risk-Informed Engineering and by the Manager, Licensing. The instruction shall then be revised at this time to remove this Section.

1.5.2 **IF** the above evaluation concludes that the process described in this instruction does not meet the requirements of, or is inconsistent with, the approved license amendment, **THEN** this instruction shall be revised accordingly and any evaluations or activities already performed shall be re-performed using the revised procedural requirements.

2.0 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.

This instruction was created and is maintained under the direction of the Risk-Informed Engineering Manager.

3.0 References

3.1 10 CFR 50.69, Risk-Informed Categorization And Treatment Of Structures, Systems And Components For Nuclear Power Reactors

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

3.3 NMP-ES-065, 10 CFR 50.69 Program

3.4 NMP-ES-065-001, 10CFR50.69 Active Component Risk Significance Insights

3.5 NMP-ES-065-002, 10CFR50.69 Passive Component Categorization

3.6 NMP-ES-065-004, Alternative Treatment Requirements

3.7 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

3.8 NMP-ES-066-001: Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities

4.0 Definitions

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

5.0 Responsibilities

5.1 The IDP is responsible for the following activities, as further detailed in NMP-ES-066:

5.1.1 Evaluating PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization for system functions and components that are presented to the IDP for review.

5.1.2 Reviewing 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. These results are presented to the IDP for review.

5.1.3 Evaluating recommended changes to categorization results resulting from changes to the plant, PRA model updates, changes to operational practices, as well as other applicable changes. These changes are presented to the IDP for review.

5.2 The cognizant Risk-Informed Application engineer is responsible for the following activities:

5.2.1 In concert with Site Management, establishing the criteria for and selecting the plant systems to be categorized.

5.2.2 Providing the PRA base case risk and results of sensitivity studies for SSCs in the system under review, as further detailed in NMP-ES-065-001.

5.2.3 Providing the results of other hazards analyses for those hazards that are not modeled in the PRA, as further detailed in NMP-ES-065-001.

5.2.4 Providing additional PRA Model insights which may influence the SSC categorization outcome.

5.2.5 Providing PRA risk insights in support of the passive risk categorization of SSCs, as further detailed in NMP-ES-065-002.

5.2.6 Providing PRA risk changes, resulting from model updates or other factors that could impact existing SSC categorizations.

5.3 Site Management is responsible for:

5.3.1 Providing input in establishing the criteria for and selecting the plant systems to be categorized

5.3.2 Providing 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.

5.4 The cognizant Licensing engineer is responsible for assessing the system under review for regulatory or commitment insights which may influence the SSC categorization outcome.

5.5 The cognizant System Engineer is responsible for the following:

- Developing system functions
- Mapping each component in the system to the system function(s) supported
- Assessing system health or equipment performance insights which may influence the SSC categorization outcome.
- Providing insights on relevant industry-related performance issues which may influence the SSC categorization outcome.
- Participating in the categorization of SSCs in the assigned system.

5.6 The Operations representative is responsible for:

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

6.0 **Procedure**

NOTES

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.

6.1 Essential Elements

6.1.1 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.

6.1.2 PRA Capability

NOTE

Additional details are provided in NMP-ES-065-001, 10CFR50.69 Active Component Risk Significance Insights.

The risk-informed categorization of SSCs in nuclear power plant applications requires the use of an appropriately detailed PRA of sound technical quality. 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.

6.1.3 Qualitative Insights

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

6.1.4 Passive (Pressure Retention) Risk of Components

> **NOTE**
> Additional details are provided in NMP-ES-065-002, 10CFR50.69 Passive Component Categorization.

The classification of components (including associated supports) having only a pressure retaining function (also referred to as passive components), or the passive function of active components, are required to undergo a separate passive risk assessment process. 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.

6.1.6 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.

6.1.7 Integrated Decision Making Panel

> **NOTE**
> Additional details are provided in NMP-ES-066-001 Integrated Decision-making Panel for Risk Informed SSC Categorization: Duties and Responsibilities.

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.

6.1.8 Training

Specific training and qualifications requirements for IDP members and designated alternates is detailed in NMP-ES-066-001. 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.

6.1.9 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 system or structure components associated with that system and possessing a unique component identification number in the Plant Data Management System (PDMS).

6.1.10 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.

6.2 Selection of Plant Systems to be Categorized

6.2.1 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.

6.2.2 Support systems (e.g., cooling systems or electrical distribution systems) should not be categorized 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.

6.3 Collecting and Assembling System Functional Information

6.3.1 Identify the system to be categorized

6.3.2 Develop a list of functions performed by the system.

6.3.2.1 All functions should be identified, 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.

6.3.2.2 Sources of information for the development of system functions include, but are not limited to, Maintenance Rule functions, design basis documents, system descriptions, Piping and Instrumentation Diagrams (P&IDs), and the Final Safety Analysis Report (FSAR).

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

6.3.4 Identify the components within the system.

6.3.4.1 Typically, this will consist of those 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.

6.3.4.2 Component information should be electronically available from PDMS and should be used to identify all active (i.e., not spared, deleted, or retired) components that are associated with the system of interest.

6.3.4.3 Piping segments should also be included in the list of components and uniquely identified

6.3.5 For each component, identify the system function(s) that the component supports.

6.3.5.1 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.

6.3.5.2 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.

6.3.5.3 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.

6.4 Collecting and Evaluating System Operational Information

6.4.1 Collect plant and industry operating experience relevant to the system or its components.

6.4.1.1 Focus on equipment failures or significant degradations and review for importance, commonality, and repeat occurrences.

6.4.1.2 Identify any SSCs that exhibit poor performance.

6.4.1.3 Summarize the evaluation for presentation to the IDP and identify any potential categorization or treatment impacts.

6.4.2 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.

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

6.5 Risk Evaluations based on PRA or Other Hazards Analyses

> **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).**

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

The process for assessing these risk hazards is detailed in NMP-ES-065-001 and is consistent with NEI 00-04, Rev. 0. This process will provide the following risk assessment results to be used as input into the overall categorization of SSCs:

- 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).

6.6 Passive Risk

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 NMP-ES-065-002. 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 NMP-ES-065-002 process will provide, 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.

6.7 Qualitative Risk Assessment

6.7.1 Qualitative Risk Assessment of System Functions

Each system function shall be categorized as HSS if one or more of the following questions is answered affirmatively. Otherwise, the function will be categorized as 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?

6.7.2 Qualitative Risk Assessment of Components

NOTE
This section excludes component passive risk, which is discussed in Section 6.6.

Components are given 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.

A component may be assigned a risk of LSS even it supports an HSS function if the failure of the component would not 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 not 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 not prevent an HSS function from being fulfilled (e.g., radiation monitors that do not have a direct diagnosis function, etc.).

Caution and conservative judgment should be exercised before such allowances can be taken and the associated justification should be documented.

6.8 Overall Risk Assessment of Components

Each component shall be preliminarily identified as HSS if any of the following assessments indicate that it should be HSS. Otherwise, it shall be preliminarily identified as LSS

6.8.1 Evaluation results for modeled hazards (from section 6.5 and NMP-ES-065-001)

6.8.2 Evaluation results for non-modeled hazards (from section 6.5 and NMP-ES-065-001)

6.8.3 Passive risk (from section 6.6 and NMP-ES-065-002)

6.8.4 Qualitative risk (from section 6.7.2)

6.9 Defense in Depth Assessment

For components whose overall risk is LSS from Section 6.8, an additional evaluation is required to assess the role of the components in preserving defense-in-depth related to core damage, large early release and long term containment integrity. Details on the methodology for performing the defense-in-depth assessment is provided in Attachment 1.

If the defense-in-depth assessments for either core damage or containment integrity cannot confirm the low safety significance of a particular component, then the component shall be preliminarily re-categorized as HSS. Otherwise, it remains preliminarily LSS.

6.10 Compilation of Risk Evaluation Data for IDP Presentation

For the selected system and its associated components, the following data shall be compiled, as applicable:

6.10.1 Licensing commitment review

6.10.2 Qualitative risk results for system functions

6.10.3 Operating experience review

6.10.4 Assessment of system health and equipment performance

6.10.5 PRA individual model and integrated risk assessments for modeled components

6.10.6 Evaluation results for non-modeled hazards

6.10.7 Results of PRA sensitivity studies for any of the PRAs used

6.10.8 PRA LSS components that are in the buffer zone

6.10.9 Passive risk for applicable components

6.10.10 Qualitative risk results for system components

6.10.11 Defense-in-depth assessments

6.11 IDP Evaluation of Risk Results

The IDP shall evaluate all of the available risk results and other system information and develop a consensus on the risk categorization of the system functions and components using the following guidance.

6.11.1 General Considerations

6.11.1.1 The intent of the IDP review is to ensure that SSCs have been appropriately categorized with a documented supporting basis.

6.11.1.2 The IDP may request personnel with additional expertise or information be present at the meeting to facilitate completion of the categorization.

6.11.1.3 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, all components supporting the function are initially considered to be HSS.

6.11.1.4 If a detailed categorization is performed after the initial categorization, this is a separate review by the IDP. This same criteria as the initial categorization is applied.

6.11.1.5 For HSS SSCs, if the categorization is appropriate, the IDP cannot move the SSC to an LSS category.

6.11.2 System Functions

6.11.2.1 System Functions should completely describe the system.

6.11.2.2 System Functions should be categorized in a sound, consistent, and well documented manner.

6.11.2.3 The answer to each essential question should be supported by an appropriate basis.

6.11.2.4 The IDP should ensure that the answers are reasonable and consistent, both within the selected system and, as other systems are categorized, across systems.

6.11.3 The PRA results for modeled components should be understood, 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 NMP-ES-065-001.

6.11.4 Evaluation results for non-modeled hazards (e.g., seismic risk) should be understood 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.

6.11.5 Sensitivity results should be understood including the base and integral risk for each hazard

6.11.6 Passive risk results should be understood with respect to assumptions and use of bounding assessments

6.11.7 Qualitative risk results for components should be evaluated 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)

6.11.8 Defense in depth and safety margins considerations for safety related LSS components should be confirmed 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.

6.11.9 Review of Non-Safety Related but Important-to-Safety LSS components

For non safety related LSS SSCs that are classified as important-to-safety, the IDP must consider if the risk information used in the categorization process provides an adequate basis for categorizing the SSC as LSS. 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. If the IDP concludes that the categorization of the SSC as LSS is not justified, then the IDP can re-categorize the SSC to HSS. In doing so, however, the attributes of the SSC should be identified to assure that any core damage prevention and mitigation attributes that the IDP felt were significant are included in future treatment including beyond design basis functions used in the PRA.

6.12 Blending of Risk Results and Overall Assessment

After evaluating the above results, the IDP will reach consensus on the overall categorization of the system functions and components, subject to the following:

6.12.1 A component that has been identified as HSS by the passive risk assessment must be categorized as HSS, regardless of any other factors.

NOTE

For components that have both an active and a passive function, the overall risk of the component will of course be the higher of the two. However, 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 6.13. The following criteria generally involve the active risk.

6.12.2 A component that has been identified as HSS by the PRA <u>integrated</u> risk assessment **MUST** be categorized as HSS, regardless of any other factors.

6.12.3 A component that has been identified as HSS by one or more of the non-modeled hazards evaluations **MUST** be categorized as HSS, regardless of any other factors.

6.12.4 The qualitative risk of system functions **OR** components may be revised from LSS to HSS based on IDP judgment. Conversely, the qualitative risk of components may be revised, 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 6.7.2.

6.12.5 For components that are still LSS, the results of the sensitivity studies shall be evaluated to determine if the component risk should be increased to HSS.

6.12.6 For components that are still LSS, the risk should be increased to HSS **IF** the results of defense-in-depth assessments point to a risk of HSS, **UNLESS** a justification can be made, documented, and accepted by the IDP that the risk should not be increased.

6.12.7 For components that are still LSS and in the PRA buffer zone (i.e., within 10% of the HSS threshold), the IDP should consider increasing the risk to HSS.

6.13 Component Critical Attributes

6.13.1 For those components categorized as HSS, the attributes of the component that are associated with its safety significance should be reviewed by the IDP. Typically, such attributes are developed 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, the critical attribute(s) would include, but not necessarily be limited to, pressure retention.

6.13.2 For those components supporting HSS functions but categorized as LSS based on mitigating factors, the attributes of the component that are associated with supporting the HSS functions should be documented as critical, with the clarification that loss of the attribute would not, in and of itself, fail the function.

6.14 Final Classification

The IDP will 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

The results of the final classification of SSCs will be documented as detailed in Section 7.

6.15 Periodic Reviews and Performance Feedback

6.15.1 Periodic reviews shall be conducted to ensure continued validity and performance monitoring for those SSCs that have been categorized. In support of this, the periodic reviews should:

- Be 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.

6.15.2 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, an immediate evaluation and review should be performed prior to the normally scheduled periodic review.

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

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

6.15.4 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.

6.15.5 The IDP will 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

6.15.6 The IDP has the final decision regarding re-categorizations.

6.16 Critical Changes

NOTE

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.

6.16.1 A critical change occurs whenever the risk of a safety-related component changes from LSS to HSS. Components that have not had any alternative treatments applied are not subject to critical changes. Critical changes do not apply to increases in the risk of system functions, however, such changes can result in a critical change at the component level.

6.16.2 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.

6.16.3 As soon as the potential for a critical change is identified, a Condition Report will be initiated, in accordance with the Corrective Action Program. The Condition Report SHALL include the necessary data to support a proper evaluation. At a minimum, the following actions will be generated for the Condition Report.

NOTE

If conditions/events do not permit the below timeframes to be satisfied, the Integrated Working Group Chairman shall ensure that interim compensatory measures are instituted until the next required action can be accomplished.

6.16.3.1 The IDP will convene to determine the appropriateness of the potential change within 14 calendar days of the initiation of the Condition Report action.

6.16.3.2 If an electronic database is being used to provide RISC classifications for use by the Plant, the database shall be revised to reflect the new RISC-1 classification within 14 days of IDP approval of the change.

6.16.3.3 The Risk Basis Document for the applicable system shall be amended to reflect the new RISC-1 classification within 30 days of IDP approval of the change.

6.16.3.4 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. License compliance and operability must be considered as necessary.

6.16.3.5 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. Individual actions will be assigned to each owner to complete the assessment. A list of Alternative Treatment Programs can be found in NMP-ES-065-004.

6.16.4 In the unlikely event that the IDP decides that the critical change is not valid, the owners of the associated Condition Report action items will be notified as soon as possible, the action items will be revised to incorporate the decision of the IDP, and any changes will be rescinded, as applicable.

7.0 <u>Records</u>

7.1 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. Typically, this documentation should consist of the following:

- 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

7.2 Documents generated by this instruction are considered QA records and shall be stored using the following R type in the **Corporate doc base**.

7.2.1 After the IDP approves categorization results of a system, the results will be captured in a Risk Based Document (RBD). The RBD will also contain associated supporting information that was used to categorize the system. The RBD will reside in the Corporate doc base, and the Corporate R Type is PRA05.017.

7.2.2 The IDP meeting minutes shall be stored per NMP-ES-066-001.

7.3 A suitable plant-wide electronic means of providing the RISC classifications of components should be implemented. This data is to be updated to reflect categorization data changes within a reasonable period of time, notwithstanding the specific time constraints associated with critical changes.

7.4 The RBD should be updated to incorporate changes to categorization data, if applicable, at least at the same frequency as the scheduled Periodic Review for the associated system. This update will take place 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.

7.5 The RBD may also be updated through the use of an amendment-type change process. Outstanding amendments will be incorporated through a general revision on at least the same frequency as the scheduled Periodic Review for the associated system.

8.0 <u>Commitments</u>

None

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 **<u>without</u>** 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 the LSS categorization of the function/SSC has been confirmed.
- If the result is in the region entitled "Potentially Safety-significant," then the function/SSC should be classified as 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 the SSC remains Candidate LSS for the IDP.

For example, if a BWR found that the low pressure core spray (LPCS) system pumps were LSS in the categorization process using risk information, then their categorization would be confirmed using Figure 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
  - not locked closed or only locally operated?
- Does the SSC support containment isolation for containment penetrations that are:
  - Part of the reactor coolant system pressure boundary, and
  - > 3/8" in diameter, and
  - not locked closed or only locally operated?

<u>Early Hydrogen Burns</u>

- 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 the SSC remains 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.

Figure 1

DEFENSE-IN-DEPTH MATRIX

Frequency	Design Basis Event	≥3 diverse trains OR 2 redundant systems	1 train + 1 system with redundancy	2 diverse trains	1 redundant automatic system
>1 per 1-10 yr	Reactor Trip Loss of Condenser				
1 per 10-10^2 yr	Loss of Offsite Power Total Loss of Main FW Stuck Open SRV (BWR) MSLB (outside cntmt) Loss of 1 SR AC Bus Loss of Instr/Cntrl Air				POTENTIALLY SAFETY SIGNIFICANT
1 per 10^2-10^3yr	SGTR Stuck Open PORV/SV RCP Seal LOCA MFLB MSLB Inside Loss of 1 SR DC bus	LOW SAFETY SIGNIFICANCE CONFIRMED			
<1 per 10^3 yr	LOCAs Other Design Basis Accidents				

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 5

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

Instruction Owner: Paul Hayes / Fleet Engineering Services Director / Corporate
(Print: Name / Title / Site)

Approved by: ______________________________
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:

Corporate	FNP	HNP	VEGP 1&2	VEGP 3&4
N/A		N/A		N/A

Writer(s):
Vish Patel
Stephanie Agee

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	**NONE**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on site for reference as needed.	**ALL**

Version Number	Version Description
1.0	Initial Issue

Table of Contents

1.0 Purpose

This procedure establishes the concepts of the Integrated Decision-making Panel (IDP) for the Risk Informed Tech Spec Initiative 5b process (Surveillance Frequency Control Program (SFCP)) (for which specifics are described in NMP-ES-066-001) and 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 SFCP and 50.69 processes, respectively.

2.0 Applicability

This procedure is applicable to the 50.69 and SFCP processes 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 NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies Industry Guidance Document", revision 1

3.4 NMP-ES-065, 10CFR50.69 Program

3.5 NMP-ES-072, Surveillance Frequency Control Program

3.6 NMP-ES-066-001, Integrated Decision-Making Panel for Surveillance Frequency Control Program (SFCP) Duties and Responsibilities

3.7 NMP-ES-066-002, Integrated Decision-Making Panel for Risk Informed SSC Categorization Duties and Responsibilities

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.

4.1.2 SFCP IDP - the IDP convened to review changes to surveillance test intervals under the SFCP.

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 Surveillance Frequency Control Program and the IDP Chairperson are defined in NMP-ES-066-001.

5.3 The responsibilities of the site IDP for the 10CFR 50.69 Categorization Process and the IDP Chairperson are defined in NMP-ES-066-002.

5.4 Risk Informed Engineering Department

5.4.1 Ensures that training is developed for the IDPs

5.4.2 Ensures IDP members receive the appropriate training before participating in IDP deliberations

5.5 Site Operations Manager or designee

5.5.1 Selects individuals to serve as IDP members

5.5.2 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 the specific IDP they are part of.

6.1.3 The site IDP is envisioned as a group that collectively meets the requirements of both the SFCP and 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. A site IDP convened to review 50.69 material shall **NOT** review an SFCP evaluation. Likewise, the site IDP convened for review of SFCP material shall **NOT** review 50.69 packages.

6.1.8 The IDP reviews the material presented to it and makes a decision whether to approve the material/change (in the case of SFCP) or the recommended HSS/LSS categorization (in the case of 50.69). 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 <u>Records</u>

7.1 Records related to the Surveillance Frequency Control Program are defined in NMP-ES-066-001.

7.2 Records related to the 10 CFR 50.69 Categorization Process are defined in NMP-ES-066-002.

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 6

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

Instruction Owner: ______________________________
(Print: Name / Title / Site)

signed by: on:

Approved by: ______________________________
(Peer Team Champion/Procedure Owner's Signature / Date)

Effective Dates:

Corporate	FNP	HNP	VEGP 1&2	VEGP 3&4
N/A				

Writer(s):

PRB review is required for this instruction.

PROCEDURE USAGE REQUIREMENTS		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	**NONE**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on site for reference as needed.	**ALL**

Version Number	Version Description
1.0	Initial Issue

Table of Contents

1.0 Purpose

1.1 This procedure establishes the Integrated Decision-making Panel (IDP), and defines its structure, responsibilities, and qualifications. It addresses the IDPs for the 10CFR50.69 risk informed categorization (50.69) process only.

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

- NMP-ES-065, 10CFR50.69 Program
- NMP-ES-065-003, 10CFR50.69 Risk Informed Categorization for Structures, Systems, and Components
- NMP-ES-065-001, 10CFR50.59 Active Component Risk Significance Insights
- NMP-ES-065-002, 10CFR50.69 Passive Component Categorization
- NMP-ES 065-004, Alternative Treatment Requirements
- 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), which are discussed separately in instruction NMP-ES-065-004.

1.4 50.69 IDP Description

An 50.69 IDP serves as a multi-disciplinary review group for the risk informed categorization process.

2.0 Applicability

This document is applicable to all sites having an NRC approved 10CFR50.69 Risk Informed Categorization program.

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-065-003, Risk Informed Categorization and Treatment of Systems, Structures, and Components instruction

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

4.0 Definitions

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.

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 The site IDP has the following responsibilities for which detailed guidance is provided in NMP-ES-065-003:

5.1.1 Evaluating PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization for system functions and components that are presented to the IDP for review.

5.1.2 Reviewing 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.

5.1.3 Evaluating recommended changes to categorization resulting from changes to the plant, PRA model updates, changes to operational practices, as well as other applicable changes.

5.2 The site IDP chairperson has the following responsibilities

5.2.1 Schedule and run the site IDP meetings.

5.2.2 Ensure that quorum requirements are met for IDP meetings.

5.2.3 Ensure site IDP meeting minutes are prepared.

5.2.4 Ensure site IDP meeting minutes are approved.

5.3 The site IDP secretary (Risk Informed Engineering department) has the following responsibilities

5.3.1 Ensure that minutes of site IDP meetings are retained along with other required IDP records per site QA records process.

5.3.2 Forward the site IDP meeting minutes to the corporate IDP oversight committee secretary

5.3.3 Facilitate qualification of IDP members.

5.3.4 Notify the site IDP Chairperson when 50.69 IDP meeting is needed.

6.0 Procedure

6.1 50.69 IDP Organization

6.1.1 Composition

6.1.1.1 The 50.69 IDP shall be composed of members covering the following functional areas:

a) Operations (SRO)
b) Safety Analysis
c) Design Engineering
d) Systems Engineering
e) Probabilistic Risk Analysis (PRA)

6.1.1.2 The site 50.69 IDP should include members from the following organizations

a) Site Operations (SRO)
b) Safety Analysis
c) Site Design Engineering
d) Site System Engineering

e) Site Risk Informed Application

f) Site Nuclear Licensing

g) Site Maintenance

6.1.1.3 The Operations Manager (or designee) selects primary and alternate members to serve on the IDP.

6.1.1.3.1 The qualified alternate(s) are designated to sit on the panel for absent member(s).

6.1.1.3.2 The Operations Manager (or designee) will act as a Chairperson.

6.1.2 Quorum

6.1.2.1 A Quorum for the 50.69 IDP shall consist of at least five qualified persons collectively having site specific expertise in the functional areas listed in 6.1.1.1.

6.1.3 Qualifications

6.1.3.1 All members need to have:

a) A good understanding of PRA concepts and the analyses performed for risk informed categorization.

b) A good understanding of the risk informed categorization process.

c) A good understanding of the risk informed categorization requirements.

d) Experience with the specific plant being evaluated.

6.1.3.2 All IDP members must have completed an IDP member qualification form.

6.1.3.3 It is suggested that a primary and alternate member be qualified in each functional area

6.1.4 IDP Training

6.1.4.1 Initial IDP Training for the 50.69 IDP shall include:

a) The purpose of risk informed categorization including exempted regulations for low safety significance SSCs.

b) The categorization process

c) Risk informed defense in depth philosophy and how it is maintained.

d) Details of the IDP process including roles and responsibilities

e) PRA fundamentals pertinent to the 50.69 program

f) 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)

6.1.4.2 Refresher training should be provided to IDP members every 3 years.

6.1.4.3 Initial training shall be documented using the form NMP-ES-066-001-F01.

6.2 Functions

6.2.1 50.69 IDP Meetings

6.2.1.1 The 50.69 IDP should meet when any of the following apply.

6.2.1.1.1 When a risk categorization is completed in accordance with NMP-ES-065-003 and ready for IDP review.

6.2.1.1.2 When plant or PRA changes require re-evaluation of categorization.

6.2.1.1.3 To review the results of the periodic review of the program.

6.2.1.1.4 As convened by the chairperson.

6.2.1.2 Meetings shall not be conducted without a quorum present.

6.2.1.3 For scheduled IDP meetings, an effort should be made to have all primary members present.

6.2.1.3.1 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.

6.2.2 Minutes of Meetings

6.2.2.1 The IDP Chairperson will ensure the minutes of IDP meetings are prepared.

6.2.2.2 At a minimum, the minutes will 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.

See attachment 1 for example meeting minute format.

6.2.2.3 The minutes will be numbered sequentially for each calendar year, reviewed by the members, and approved by the Chairperson.

6.2.2.3.1 The minutes for each meeting should be prepared, reviewed and approved within 30 days of the meeting.

6.2.2.3.2 A copy of the minutes will be forwarded to the Operations Manager for review.

6.2.2.3.3 The meeting minutes shall be retained as a quality record. The site IDP secretary will ensure that the meeting minutes are stored per site QA records process. Fleet R Type is RR5.018.

6.2.2.3.4 The site IDP secretary will also forward the meeting minutes to the corporate IDP oversight committee secretary.

7.0 Records

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

The R type is for fleet.

8.0 Commitments

None

Attachment 1

50.69 IDP Meeting Minutes Template

Meeting number: 20__-__ **Meeting Date:** **Page of**

MEETING CONVENED: **AM/PM** **MEETING ADJOURNED:** **AM/PM**

THIS MEETING CHAIRED BY:

( ) Chairperson ______________________

 # MEMBERS PRESENT:

 # Operations

 # Design engineering

 # Safety Analysis

 # System Engineering

 # PRA

Nuclear Licensing

Maintenance

 # required for quorum

*** Denotes part-time attendance** **** Denotes phone or video attendance**

Summary of Meeting Minute Contents

Categorization packages (List system Numbers)	Periodic monitoring	Periodic or unplanned reivew

MEETING NO.: 20_-- __ **DATE:** ____________ ; **PAGE** ___ **OF** ___

Minutes:

____

____

THESE MINUTES APPROVED IN IDP MEETING NO.: 20_-- ___ **ON** ____________

Date

____

IDP CHAIRMAN'S SIGNATURE/Date

____

IDP SECRETARY'S SIGNATURE/Date

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request
Draft Risk-Informed Categorization Procedures

Enclosure 7

Draft NMP-ES-066-002-F01
Risk Informed Categorization Integrated Decision Making Panel
Qualification Form – 50.69

RISK INFORMED CATEGORIZATION INTEGRATED DECISIONMAKING PANEL (IDP) TRAINING/QUALIFICATION RECORD for ____________________(site)

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 = 10CFR50.69 Program

NMP-ES-65-003 = Risk Informed Categorization and Treatment of Systems, Structures, and Components instruction

NMP-ES-065-001 = 10CFR50.69 Active Component Risk Significance Insights Instruction

NMP-ES-065-002 = 10CFR50.69 Passive Components Categorization Instruction

NMP-ES-065-004 = Treatment

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:

[ ] IDP Chairperson

[ ] Plant Operations

[ ] Design Engineering

[ ] System Engineering

[ ] Probabilistic Safety Assessment

[ ] Safety Analysis

[ ] Licensing

[ ] Maintenance

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

[ ] Operations

[ ] Design Engineering

[ ] Systems Engineering

[ ] Probabilistic Risk Analysis

[ ] Safety analysis

[ ] Licensing

[ ] Maintenance

Manager of Department Represented: ______________________ Date: ________

Site IDP Chairperson: ______________________ Date: ________

When approved the IDP Chairperson shall submit this form to the training coordinator for submittal to Training Records location.