(17)

DOCKET NUMBER
PROPOSED RULE PR 50
(68FR 26511)

DOCKETED
USNRC

August 29, 2003 (11:31AM)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Project Number 694

August 28, 2003

WOG-03-444

Domestic Members
AmerenUE
Callaway
American Electric Power Co.
D.C. Cook 1 & 2
Arizona Public Service Co.
Palo Verde 1, 2 & 3
Constellation Energy Group
Calvert Cliffs 1 & 2
Dominion Nuclear Connecticut
Millstone 2 & 3
Dominion Virginia Power
North Anna 1 & 2
Surry 1 & 2
Duke Energy
Catawba 1 & 2
McGuire 1 & 2
Entergy Nuclear Northeast
Indian Point 2 & 3
Entergy Nuclear South
ANO 2
Waterford 3
Exelon Generation Company LLC
Braidwood 1 & 2
Byron 1 & 2
FirstEnergy Nuclear Operating Co.
Beaver Valley 1 & 2
FPL Group
St. Lucie 1 & 2
Seabrook
Turkey Point 3 & 4
Nuclear Management Co.
Kewaunee
Palisades
Point Beach 1 & 2
Prairie Island
Omaha Public Power District
Fort Calhoun
Pacific Gas & Electric Co.
Diablo Canyon 1 & 2
Progress Energy
H. B. Robinson 2
Shearon Harris
PSEG – Nuclear
Salem 1 & 2
Rochester Gas & Electric Co.
R. E. Ginna
South Carolina Electric & Gas Co.
V. C. Summer
Southern California Edison
SONGS 2 & 3
STP Nuclear Operating Co.
South Texas Project 1 & 2
Southern Nuclear Operating Co.
J. M. Farley 1 & 2
A. W. Vogtle 1 & 2
Tennessee Valley Authority
Sequoyah 1 & 2
Watts Bar 1
TXU Electric
Commanche Peak 1 & 2
Wolf Creek Nuclear Operating Corp.
Wolf Creek

International Members
Electrabel
Doel 1, 2, 4
Tihange 1 & 3
Electricité de France
Kansai Electric Power Co.
Mihama 1
Takahama 1
Ohi 1 & 2
Korea Hydro & Nuclear Power Co.
Kori 1 – 4
Ulchin 3 & 4
Yonggwang 1 - 5
British Energy plc
Sizewell B
NEK
Krško
Spanish Utilities
Asco 1 & 2
Vandellos 2
Almaraz 1 & 2
Ringhals AB
Ringhals 2 – 4
Taiwan Power Co.
Maanshan 1 & 2

Annette L. Vietti-Cook, Secretary
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Rulemaking and Adjudication Staff

Subject: Westinghouse Owners Group
Westinghouse Owners Group Comments and Feedback on the Proposed Rulemaking for 10 CFR Part 50.69 "Risk Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

The Westinghouse Owners Group (WOG) has reviewed the Commission's Notice of Proposed Rulemaking for 10 CFR Part 50.69 "Risk Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" as published in the Federal Register on Friday, May 16, 2003. The primary review effort was carried out by a team of Westinghouse and utility personnel who have been intimately involved in the development of the industry guidance related to this rulemaking, including the pilot plant efforts to validate and improve the guidance. In addition, utility members of the WOG in a wide range of disciplines that would eventually be impacted by the new rule were invited to provide their input. The comments and feedback contained in this letter and its attachments are the culmination of that intensive review effort. We would like to thank the Commission for the opportunity to provide comments and feedback at this time.

The WOG would like to commend the Commission on the initiative to risk-inform the treatment requirements for structures, systems and components that has culminated in the publication of this proposed rule. We believe that this is a very large step forward in achieving the overall goal of the Commission to include risk information and insights into the design and operation of commercial nuclear power plants with the intent of further improving safety and, at the same time, removing burdensome

Template = SECY-067

SECY-02

regulatory requirements. We agree with most of the provisions of the proposed rule and our comments primarily deal with the implementation of the new requirements of the proposed rule.

Of primary concern to the WOG members is the appropriateness and clarity of the information provided in the Statement of Considerations (SOC) accompanying the proposed rule. Our understanding of the purpose of the SOC is to provide additional information to clarify the intent of the rule language. As such , the SOC should be limited to considerations that are important to understanding the background assumptions and processes that led to the rule language. This is not the case with the SOC accompanying the proposed rule as published in the Notice of Proposed Rulemaking. The SOC contains significant examples of Commission expectations, requirements, and other guidance that, in many cases, is not consistent with the rule language or intent as proposed. As such, the SOC represents considerable uncertainty in the regulatory process to the extent that, if retained in the publication of the final rule, may result in the refusal of licensees to implement this voluntary rule.

Detailed comments and feedback are provided in the attachment to this letter. The comments and feedback in the attachment are presented in five categories: a) the requirements in rule itself, b) appropriateness and clarity of the information in the Statement of Considerations, c) technical information in the Statement of Considerations and the proposed rule, d) editorial comments, e) feedback on the specific areas where input was requested, and f) the draft Regulatory Guide DG-1121. If you have any questions related to these comments, please contact Mr. Maurice Dingler, Chairman of the WOG Systems and Equipment Subcommittee at 620-364-4127.

Very truly yours,

Robert H. Bryan, Chairman
Westinghouse Owners Group

WOG-03-444
August 28, 2003

cc: WOG Steering Committee
WOG Management Committee
WOG Licensing Subcommittee
WOG Systems & Equipment Engineering Subcommittee
WOG Risk Management Subcommittee
D. Holland, USNRC OWFN 07 E1 (1L, 1A)
Project Management Office
M.A. Ahmed
G.A. Brassart
C.B. Brinkman
J.A. Brown
K.R. Balkey
R.J. Lutz
H.A. Sepp

INDEX TO WOG COMMENTS

A. Rule Requirements

B. Appropriateness of the Statement of Considerations

C. Technical Information in the Statement of Considerations

D. Editorial Comments

E. Specific Areas of Requested Input

F. Draft Regulatory Guide DG-1121

A. RULE REQUIREMENTS

A.1 CRITERION FOR UNISOLATED CONTAINMENT

The selection of a 1 inch penetration size as a bounding criteria for all light water reactors is inconsistent with other regulatory documents related to large early releases. For example, the most recent regulatory guidance on containment penetration size that is risk significant is provided in draft Appendix H - Containment Integrity Significance Determination Process (SDP) that was provided at the NRC's "ROP MONTHLY PUBLIC MEETING, MAY 22 2003". In that document and in the excerpted piece from Federal Register Page 16525, Column 3 (above), risk significance is equated to LERF and to a 100 volume percent per day leakage rate from containment. According to that document at page 45, "The 100 volume percent per day leakage rate is approximately equivalent to a hole size in containment of 2.5 to 3 inches in diameter for PWRs with large dry containments, 2 inches for PWRs with ice condenser containments, 1 inch for BWRs with Mark I or II containments, and 2.5 inches for BWRs with Mark III containments."

We recommend that the size of the line / penetration (or leakage rate) not be specified in the rule language in order to facilitate reasonable changes to that size that might be indicated by new information or analyses in the future. In addition, the Statement of Considerations should be revised to be consistent with other regulatory documentation (i.e., Appendix H of the SDP guidance).

Applicable NOPR Federal Register Locations:

Federal Register Page 26525	Column 1	Bottom Quarter of Column

"2. Penetrations less than 1 inch in equivalent diameter."

and Federal Register Page 26525	Column 2	Top of Column

"4. The valve size is 1-inch nominal pipe size or less."

and Federal Register Page 26525	Column 2	Middle of Column

"The second criterion for reducing the scope of Type B testing *(i.e.,* penetrations less than 1 inch in equivalent diameter) is essentially the same as the fifth criterion for reducing the scope of Type C testing *(i.e.,* valve size is 1-inch or less). By definition penetrations of this size do not contribute to large early release."

and Federal Register Page 26525	Column 3	First Full Paragraph

"NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' dated September 1995, calculated the containment leakage necessary to cause a significant increase in risk and found that the leakage rate must typically be approximately 100 times the Technical Specification leak rate, La. It is improbable that even the leakage of multiple v valves in the categories under consideration would exceed this amount."

and Federal Register Page 26549	Column 3	50.69(b)(1)(ix)(A) and 50.69(b)(1)(ix)(B)(4)

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

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

A.2 TREATMENT REQUIREMENTS FOR EQ AND SEISIMIC

The rule contains requirements for environmental qualification of RISC-3 SSCs that can be interpreted to be quite similar to the requirements of 50.49 and Part 100 Appendix A. The level of detail of requirements in this section is greater than in the categorization section, while the categorization is more important to the potential for reduction in treatment for low safety significant SSCs permitted by this rule. It is our belief that the rule language can be simplified by deleting the parenthetical clarifications for environmental and seismic conditions without changing the intent of the treatment requirements for design control. As the rule language for this section now stands, the requirements for RISC-3 SSCs can exceed the requirements presently imposed on RISC-1 SSCs for some plants.

Applicable NOPR Federal Register Locations:

Federal Register Page 26550 Column 31 50.69(d)(2)(i)

"(i) Design control. Design functional requirements and bases for RISC-3 SSCs must be maintained and controlled. RISC-3 SSCs must be capable of performing their safety-related functions including design requirements for environmental conditions (i.e., temperature and pressure, humidity, chemical effects, radiation and submergence) and effects (i.e., aging and synergism); and seismic conditions (design load combinations of normal and accident conditions with earthquake motions);"

A.3 EXTENT OF NRC APPROVAL OF LICENSEE SUBMITTALS

The rule is ambiguous concerning extent of implementation of 50.69 to systems other than those specifically referenced in the required license amendment. It is our understanding that the approval of the license amendment submittal by the Commission would permit licensees to use the categorization and high level treatment processes described in the license amendment request for additional systems without the need to obtain additional approvals from the Commission. The rule language should be revised to clarify that only the initial implementation of 50.69 needs to be approved by the Commission using a license amendment request.

Applicable NOPR Federal Register Locations:

Federal Register Page 26550 Column 1 50.69(b)(3)

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

B. APPROPRIATENESS OF THE STATEMENT OF CONSIDERATIONS

B.1. TREATMENT OF RISC-3 SSCS

One area of the Statement of Considerations that we believe needs significant attention by the Commission in developing the final rule is the Statement of Considerations sections on Treatment for SSCs categorized under the 50.69 process. The rule requirements for treatment (i.e., 50.69(d)) were developed to include the high level attributes to be applied to the SSCs categorized according to the requirements in 50.69(c) while leaving flexibility for licensee implementation. A performance-based approach is specified in 50.69(e) to provide assurance that adequate treatment is being applied to those SSCs. However, the Statement of Considerations (Section V.5.0) includes significant details that the Commission expects of licensees in

developing the plant specific treatment requirements for categorized SSCs. While the Commission is careful to state that these are expectations and that the failure to implement a specific expectation is not by itself a regulatory concern, these expectations represent a significant uncertainty to licensees. While these expectations are not part of the text of the 50.69 rule, their inclusion in the Statement of Considerations carries significant weight in determining the Commissions intent.

We believe that this is inappropriate material to include in the Statement of Considerations for a number of reasons:

- The detailed expectations and examples do not clarify the intent of the treatment requirements in the rule;
- It defeats the rule objective of assuring that the categorization process is robust and therefore subsequent treatment can be performance based; and
- The important elements of treatment may change as experience is gained with 50.69 and inclusion of detailed examples in the Statement of Considerations may no longer be appropriate to understand the intent of the rule language.

We believe that a robust categorization process, supported by the Integrated Decision-Making Panel deliberations, which includes PRA as one consideration, provides an adequate foundation for reducing the treatment requirements for low safety significant SSCs. Therefore, only the high level attributes of treatment for RISC-3 SSCs needs to be discussed in this rulemaking as represented in the SOC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26541 to 26545 All of Section V.5.2

Examples include:

- At page 26543, fist column: "Qualification testing of an SSC would be necessary if no suitable alternative method is available for showing that the SSC will perform its design basis function during an earthquake."
- At page 26543, second column: "In addition to appropriately specifying in the procurement the desired component, the licensee/applicant would also be expected to conduct activities upon receipt to confirm that the received component is what was ordered."
- At page 26543, third column: "Testing—Testing under simulated design basis conditions could be performed on the SSC."
- At page 26544, first column: "To determine that SSC will remain capable until the next scheduled activity, a licensee would have to obtain sufficient operational information or performance data to provide reasonable confidence that the RISC–3 pumps and valves will be capable of performing their safety function if called upon to function under operational or design basis conditions over the interval between periodic testing or inspections."

B.2. TREATMENT OF RISC-1 AND RISC-2 SSCS

The SOC contains requirements for RISC-1 and RISC-2 SSCs that appear to require that additional treatment be applied and that their performance be more closely monitored than the currently required. We believe that this is an unnecessary step. The treatment and performance of RISC-1 and RISC-2 SSCs is already closely scrutinized under existing regulatory requirements (e.g., 50.65) and licensee programs (e.g., Corrective Action Programs). Also, imposing additional requirements for RISC-1 and RISC-2 SSCs in systems where 50.69 is implemented by a licensee (as opposed to safety related SSCs in systems where 50.69 is not implemented at the same plant) would create an uneven treatment of safety related SSCs at each plant. Any additional treatment requirements for RISC-1 and RISC-2 SSCs should be removed from the SOC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26513 Column 3 1st Partial Paragraph

"As part of this process, those SSCs found to be of risk-significance would be brought under a greater degree of regulatory control through the requirements being added to the rule designed to maintain consistency between actual performance and the performance considered in the assessment process that determines their significance."

and Federal Register Page 26514 Column 1 1st Paragraph of Section III

"Under this approach, a licensee would be allowed to reduce special treatment requirements for SSCs that are determined to be of low safety significance and would enhance requirements for treatment of other SSCs that are found to be safety significant. "

and Federal Register Page 26514 Column 3 1st Full Paragraph

"The proposed rule would impose greater treatment requirements on SSCs that perform safety-significant functions (RISC–1 and RISC–2 SSCs) to ensure that defense-in-depth and safety margins are maintained."

and Federal Register Page 26517 Column 1 Bottom of Last Paragraph

"In the proposed rule, a licensee would be required to ensure that the treatment applied to RISC–1 and RISC–2 SSCs is consistent with the performance credited in the categorization process. This includes credit with respect to prevention and mitigation of severe accidents. In some cases, licensees might need to enhance the treatment applied to RISC–1 or RISC–2 SSCs to support the credit taken in the categorization process, or conversely adjust the categorization assumptions to reflect actual treatment practices."

and Federal Register Page 26540 Column 3 First Full Paragraph

"If a § 50.69 licensee chooses to categorize a selective set of SSCs as RISC–3, and the categorization of SSCs as RISC–3 is based on credit taken for the performance of other plant SSCs (that would be RISC–1 or RISC–2, whether or not these SSCs are within the selective implementation set), then the licensee must ensure that consistency of performance with what was credited in the categorization."

Federal Register Page 26540 Column 2 Section V.5.1

"To meet this, a licensee should first evaluate the treatment being applied in light of the credit being taken in the categorization process, with appropriate adjustment of treatment or categorization to achieve consistency as necessary."

B.3. FEEDBACK OF DATA COLLECTION

The assessment of data collected should be an engineering function and the decision to "feedback" into the categorization and treatment processes should not be required unless there is a significant deviation in SSC performance compared to that used during the categorization process. The words in the Statement of Considerations (page 26541) should be clarified to match the "appropriate" text from the rule language (page 26550).

Applicable NOPR Federal Register Locations:

Federal Register Page 26541 Column 2 Top of Column

"Finally, when data is collected, it must be fed back into the categorization and treatment processes,..."

and Federal Register Page 26550 Column 3 50.69 (e)(1)

"In a timely manner but no longer than every 36 months, the licensee shall review changes to the plant, operational practices, applicable industry operational experience, and, as appropriate, update the PRA and SSC categorization."

B.4. SSCS/EVENTS NOT MODELED IN THE PRA

The 5 criteria for considering the impact of an SSC and the subsequent 5 bullets that describe the factors to be considered for each criterion at page 26537 of the NOPR are regulatory guidance as opposed information that clarifies the intent of the rule language. Therefore it is inappropriate to include this in the SOC.

Additionally, the criterion are sufficiently vague as to invite interpretation issues. For example, the fifth criterion refers to monitoring of post accident conditions. Almost all plant instrumentation can be used to monitor post accident conditions. A framework for considering the importance of the parameters modeled by plant instrumentation has been developed and presented in Regulatory Guide 1.97, Rev. 3. Although these importance considerations do not include any risk insights, they provide a valuable tool in considering the relative importance of each. Those Regulatory Guide 1.97 parameters that are classified as Type 1 of Class A variables are included in the plant Technical Specifications to ensure their availability following a postulated accident because of their perceived importance as derived from design basis insights. Similarly, a large number of SSCs are included in the plant Emergency Operating Procedures based on the design of the Procedures to cover design basis and credible beyond design basis plant conditions. However, only a few of the SSCs are key to the successful recovery of from a postulated accident using the Emergency operating Procedures. We believe that these criteria should not be included in the SOC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26537 Column 2 Bottom Half of Column

"For these unmodeled events, the IDP assessment should consider whether an SSC has an impact on the plant's capability to: (1) Prevent or mitigate accident conditions, (2) Reach and/or maintain safe shutdown conditions, (3) Preserve the reactor coolant system pressure boundary integrity, (4) Maintain containment integrity, or (5) Allow monitoring of post-accident conditions. In determining the importance of SSCs for each of these functions, the following factors should be considered: • Safety function being satisfied by SSC operation • Level of redundancy existing at the plant to fulfill the SSC's function • Ability to recover from a failure of the SSC • Performance history of the SSC • Use of the SSC in the Emergency Operating Procedures or Severe Accident Management Guidelines"

C. TECHNICAL INFORMATION IN THE STATEMENT OF CONSIDERATIONS

C.1. NRC REVIEW OF PRA

The discussion of the NRC review requirements for the PRA is contradictory in the various sections of the SOC. In particular, at page 26516, the implication is that a detailed review of the PRA is necessary because there are key assumptions and models that could impact categorization results. However, at other sections of the SOC, the implication is that the review of the PRA will be at a higher level that includes how peer review findings were addressed. We believe that the appropriate level of Commission review of the PRA is that laid out in draft Regulatory Guide 1122, ""DETERMINING THE TECHNICAL ADEQUACY OF PRA RESULTS FOR RISK-INFORMED ACTIVITIES". This does not include an in-depth review of the PRA nor does it include approval of the PRA. With respect to external events (fire, seismic, etc.) and other operating modes, the NRC review of the PRA methods (either quantitative or qualitative) should be limited to the process used and the consideration of the applicability of peer review comments from the internal events PRA to these other PRAs. We would also like to point out that the PRA

is only one consideration in the deliberations of the Integrated Decision-Making Panel; the categorization is not based solely on the PRA.

Applicable NOPR Federal Register Locations:

Federal Register Page 26516 Column 1 Last Paragraph

"This review will determine whether the licensee's application satisfies the § 50.69 requirements, and consider the adequacy of the PRA, focusing on the results of the peer review and the actions taken by the licensee to address any peer review findings. The Commission has determined that a focused NRC staff review of the PRA is necessary because there are key assumptions and modeling parameters that can have a significant enough impact on the results such that NRC review of their adequacy for this application is considered necessary to verify that the overall categorization process will yield acceptable decisions."

Federal Register Page 26528 Column 3 First Full Paragraph

"While detailed regulatory guidance has been developed to provide guidance for implementing categorization consistent with the proposed rule requirements, the Commission concluded that a prior review and approval was still necessary to enable the NRC staff to review the scope and quality of the plant-specific PRA taking into account peer review results. The NRC staff would also review other evaluations and approaches to be used such as margins-type analyses."

and Federal Register Page 26528 Column 3 Last Partial Paragraph

"As discussed above, prior NRC review and approval of the licensee's proposed PRA, basis for sensitivity studies and evaluations, and results of PRA review process is required."

C.2. ELEMENTS OF 50.69 REVIEW SUBMITTAL

We agree that the scope of the NRC review and approval process for implementation of 50.69 should be limited to the categorization process, including the PRA quality per the process outlined in DG-1122. A more detailed NRC review of the PRA models should not be required to assure that the categorization process is robust. Similarly, a detailed review of the categorization of each SSC should not be required as long as the licensee follows a process that has been endorsed by the NRC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26530 Column 3 Last Paragraph

"Thus, the approach that has been developed is for a rule with the minimum elements of the categorization process defined in the rule, a requirement for NRC review and approval of the categorization process (including PRA peer review information) to be used, and detailed implementation guidance (in the form of a regulatory guide)."

C.3. SSCS/EVENTS NOT MODELED IN THE PRA

For SSCs not modeled in a quantitative PRA, candidate RISC-3 SSCs have already been determined to be low safety significant because the basis for not modeling them is that their failure does not contribute to risk, For the qualitative PRA assessments (e.g., fire, seismic, etc.) if an SSCs is a candidate for RISC-3, then the screening assessment (e.g., FIVE, SMA, etc.) should identify these SSCs as low risk significant and therefore their complete failure does not contribute to the qualitative risk results. We should rely on the fact that the qualitative PRA assessments are much more bounding than the quantitative assessments and therefore there should be no requirements to assess the impact of reduced treatment for any SSC that is not modeled in either a quantitative or qualitative PRA. Thus with respect to SSCs and events not modeled in the PRA, the licensee should not be required to provide the "basis to support that the evaluations are bounding estimates of the potential change in risk", as the basis should be that it is not modeled in the PRA.

Applicable NOPR Federal Register Locations:

Federal Register Page 26516 Column 3 First Full Paragraph

"For other SSCs (not modeled in the PRA), other types evaluations would be used to provide the basis for concluding that the potential increase in risk would be small. A licensee will need to submit its basis to support that the evaluations are bounding estimates of the potential change in risk and that programs already in existence or implemented for proposed § 50.69 can provide sufficient information that any potential risk change remains small over the lifetime of the plant."

C.4. PART 21 REPORTING REQUIREMENTS

We agree with the NRC on this issue of only applying Part 21 reporting requirements to RISC-1 SSCs. RISC-2, 3 and 4 SSCs have already been determined to be of low enough safety significance that the Part 21 reporting requirements are not crucial to the NRC's regulatory processes. However, the discussion of the basis for not including RISC-2, 3 and 4 SSCs under the Part 21 reporting requirements is very long and repetitive; this section could be reduced considerably without loosing the context of the basis.

Applicable NOPR Federal Register Locations:

Federal Register Page 26519 Column 3 Last Paragraph

"Those SSCs that are viewed as being of sufficient safety significance to require Part 21 reporting are RISC–1 SSCs."

C.5. RISC-1 AND RISC-2 DEFINITIONS

The criteria for difference between RISC-1 and RISC-2 SSCs on page 26520 are not consistent with the definition of "safety related" in 50.2. The only real difference between a RISC-1 and RISC-2 SSC is based on the definition of "safety related" in 50.2. The criteria used on page 26520 to explain the difference between RISC-1 and RISC-2 SSCs are broader than the definition in 50.2.

It should also be noted that the definition of a "basic component" on page 26522 is virtually identical to the definition of "safety related" in 50.2. The applicable SOC text should be revised to be consistent with the definition of "safety related" in 50.2.

Applicable NOPR Federal Register Locations:

Federal Register Page 26520 Column 1 First Full Paragraph

"By contrast, RISC–2 SSCs are less important than RISC–1 SSCs because they do not play a role in prevention and mitigation of design basis events *(i.e.*, the SSCs that maintain integrity of fission product barriers, that provide or support the primary success paths for shutdown, or that prevent or mitigate accidents that could lead to potential offsite exposures). They are not part of the reactor protection system or engineered safety features that perform critical safety functions such as reactivity control, inventory control and heat removal. that licensees evaluate treatment being applied for consistency with key categorization assumptions, monitor the performance of these SSCs, take corrective actions, and report when a loss of a safety-significant function occurs."

and Federal Register Page 26522 Column 3 Last Paragraph

"For the purposes of this subsection, the term ''basic component'' means a facility structure, system, component or part thereof necessary to assure— (1) The integrity of the reactor coolant pressure boundary, (2) The capability to shut-down the facility and maintain it in a safe shut-down condition, or (3) The

capability to prevent or mitigate the consequences of accidents which could result in an unplanned offsite release of quantities of fission products in excess of the limits established by the Commission."

C.6. ADDITIONAL REQUIREMENTS FOR RISC-2 SSCS

No additional regulatory controls need to be placed on RISC-2 SSCs. The categorization process assumes that the reliability is consistent with the existing treatment. Since RISC-2 SSCs may be "augmented quality" SSCs as a result of their importance in Appendix R, Station Blackout, ATWS, etc. considerations, they will be within the scope of the 50.65 Maintenance Rule. Therefore, the licensee's corrective action program will be adequate to identify and resolve any performance issues related to RISC-2 SSCs. No additional requirements should be necessary and this should be clarified in the SOC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26520 Column 2 First Partial Paragraph

"As discussed in Sections III.3 and III.5 of this SOC, the Commission is proposing that additional regulatory controls be imposed on RISC–2 SSCs to prevent their performance from degrading."

C.7. OUTDATED REFERENCE

Reference to WASH-1400 is outdated. The risk insights from the more recent NUREG-1150 assessment and the IPEs are a much more relevant reference.

Applicable NOPR Federal Register Locations:

Federal Register Page 26520 Column 2 Last Paragraph

"Moreover, over the last several years, the current fleet of power reactors have been subjected to a number of risk studies, including WASH–1400 (Reactor Safety Study), and ..."

C.8. COMMON CAUSE FAILURES

We agree that multiple simultaneous failures of RISC-2 and RISC-3 SSCs are highly unlikely, even considering common cause failures, for the reasons used to exclude them from Part 21 reporting requirements. However, this is inconsistent with the discussion in other parts of this document where common cause failures across systems are raised as an issue for consideration in the categorization process.

Cross system common cause failures are rarely modeled in the PRA due to the incorporation of safeguards against common cause failures that are incorporated into plant practices.

Therefore, the requirement to consider potential effects of common-cause interaction susceptibility, including cross-system interactions and potential impacts from known degradation mechanisms is inconsistent with the requirements of other parts of this proposed regulation and further, is unnecessary from a technical perspective.

Applicable NOPR Federal Register Locations:

Federal Register Page 26520 Column 3 Last Paragraph

"The Commission believes that the multiple simultaneous failures of either RISC–2 or RISC–3 components, in the same or in different systems, is not a concern such that Part 21 reporting is necessary. Even for components of the same type, it is not likely that the installed components are identical in terms of their specific characteristics or operating and maintenance history such that a defect would lead to simultaneous failure of multiple components at the same time."

and Federal Register Page 26539 Column 1 First Full Paragraph

"One mechanism that could lead to large increases in CDF/LERF is extensive, across system common cause failures. However, for such extensive CCFs to occur would require that the mechanisms that lead to failure, in the absence of special treatment, were sufficiently rapidly developing or are not self-revealing that there would be few opportunities for early detection and corrective action. Thus, when deciding how much to assume that SSC reliability might change, the applicant or licensee is expected to consider potential effects of common-cause interaction susceptibility, including cross-system interactions and potential impacts from known degradation mechanisms."

and Federal Register Page 26516 Column 3 Bottom of First Full Paragraph

"A licensee is required to consider potential effects of common-cause interaction susceptibility and potential impacts from known degradation mechanisms. To meet this requirement, a licensee would need to: (a) Maintain an understanding of common-cause effects and degradation mechanisms and their potential impact on RISC–3 SSCs; (b) maintain an understanding of the programmatic activities that provide defenses against common cause failures (CCFs) and failures resulting from degradation; and (c) factor this knowledge into the treatment applied to the RISC–3 SSCs."

C.9. DESIGN BASIS FUNCTIONS

The "design basis" for SSCs could be interpreted to include the special treatment requirements that 50.69 proposes to remove for RISC-3 SSCs. It is suggested that this requirement be re-worded to be consistent with other sections of the Statement of Considerations – mainly that the design basis *functions* be maintained.

Applicable NOPR Federal Register Locations:

Federal Register Page 26530 Column 1 Bottom of Page

"(2) maintaining the design basis of the facility for all SSCs, including RISC–3 SSCs as described above; "

C.10. ASSURING DESIGN BASIS FUNCTIONS

The current testing and surveillance requirements for many SSCs involves simply starting a pump or exercising a valve as a means of verifying its operability and thereby providing assurance that it can perform its design basis function. The some of the discussion regarding means to assure RISC-3 SSCs can perform their design basis function requirement implies that something more is required for a RISC-3 SSCs compared to the current requirements applied to RISC-1 SSCs.

Applicable NOPR Federal Register Locations:

Federal Register Page 26541 Column 3 Bottom of First Full Paragraph

"As an example, exercising of a valve or simply starting a pump does not provide reasonable confidence in design basis capability, will not detect service-induced aging or degradation that could prevent the component from performing its design basis functions in the future, and is insufficient by itself to satisfy the intent of the rule."

C.11. RECOVERY MODELING IN PRA

One of the three considerations identified for sensitivity studies is human error probability. However, at the bottom of the page, the discussion uses the wording recovery actions. In some PRA studies, recovery has a different meaning compared to human error probabilities. Human error probabilities are modeled in PRA assessments for all operator actions. Some of these actions are a direct result of instructions in the Emergency Operating Procedures and their modeling is relatively straight-forward, based on the accident

progression as determined by equipment and/or previous operator action successes and failures. Another class of operator actions involves recovery of previously failed equipment or functions. These are typically referred to as recovery models. This involves the modeling of both the ability to recover previously failed equipment (e.g., onsite or offsite power recovery) and the operator actions to implement the recovery. By referring only to recovery actions may result in some confusion in the industry as to the intent and scope of the required of the sensitivity analyses. The SOC should be revised to address this issue.

Applicable NOPR Federal Register Locations:

Federal Register Page 26536 Column 3 Last Full Paragraph

"As part of the demonstration of PRA adequacy, the sensitivity of SSC importance to uncertainties in the parameter values for component availability/reliability, human error probabilities, and CCF probabilities should be evaluated."

and Federal Register Page 26536 Column 3 Bottom of Column

"PRAs typically model recovery actions, especially for dominant accident sequences."

C.12. IDP CONSIDERATIONS

The discussion at page 26537 of the NOPR appears to have been extracted from an early version of the ASME Code Case N-660 that was developed for categorization of pressure boundary SSCs. There are two issues associated with its usage here: 1) based on the difficulty of use during the IDP at one of the WOG pilot applications, key changes were made to the considerations, and 2) it is difficult to use when considering active components (e.g., valve motor operators). For example, the considerations discussed in element #3 are primarily aimed at pressure boundary components and it is difficult to apply these to active components.

Also, the terminology is applied to the categorization process for active SSCs using NEI-00-04 (e.g., RISC-1 through 4), which is different from that used for passive SSC categorization using ASME Code Case N-660. For example, the discussion under element #3 refers to ASME Section III Class 2 piping of the main steam and feedwater systems being categorized as RISC-1 when the ASME Code Case N-660 uses the terminology "high safety significant".

It is recommended that the detailed considerations should be left to the licensee and provided for NRC review in the documentation of the licensee's categorization process. These considerations should therefore, be deleted from the Statement of Considerations.

Applicable NOPR Federal Register Locations:

Federal Register Page 26537 Column 3 V.4.2.2

"The IDP should assess the safety significance of these SSCs by determining if: ..."

C.13. IMPACT OF PRA UPDATES ON CATEGORIZATION

It should not be necessary for the IDP to be re-convened each time the PRA is updated to consider the impact of the PRA update on the previous categorization. This should be an engineering function to determine whether the changes are significant in terms of the IDP considerations. In some case, the engineering evaluation may recommend re-convening the IDP to consider the impact of the PRA changes on specific aspects or SSCs previously categorized by the IDP. The SOC should provide clarification on this point.

Applicable NOPR Federal Register Locations:

Federal Register Page 26540 Column 2 First Partial Paragraph

"SSC categorization shall be revisited by the licensee or applicant (through the IDP) when the PRA is updated or when the other criteria used by the IDP are affected by changes in plant operational data or changes in plant design or plant procedures. Requirements for PRA updating are contained in § 50.69(e)(1)."

C.14. INTERACTION WITH 50.36 REQUIREMENTS

The SOC should clarify any possible relationship between the categorization of SSCs as safety significant and the requirements in 50.36(c)(2)(ii) Criterion 4. Specifically, we are concerned that RISC-1 and RISC-2 SSCs may incorrectly be interpreted to meet this requirement for inclusion in the plant Technical Specifications. There are presently a large number of SSCs that are safety related that are not included in the plant Technical Specifications based on considerations developed and mutually agreed to between the Commission and the industry as reflected, for example in NUREG-1431. The SOC should address this point to avoid any uncertainty or future misinterpretations of the intent of this rule.

Applicable NOPR Federal Register Locations:

Federal Register Page 26526 Column 2 Section III.4.9.2

"First, risk-informed criteria have already been established in § 50.36 for determining which SSCs should have TS requirements. Improved standard TS have already resulted in relocation of requirements for less important SSCs to other documents. Further, other improvement efforts are underway that could be implemented by individual licensees to make their plant-specific requirements more risk-informed. Thus, no changes to this rule (or its implementation) are necessary as part of § 50.69 to make the TS risk-informed or to accommodate the revised requirements of this proposed rule."

C.15. INTERACTION WITH 50.44 REQUIREMENTS

We support a review of the final rulemaking results for 50.44 to determine if the new 50.44 should be included in the applicable regulations that can be affected by the proposed 50.69 rule. The basis for inclusion in the scope of 50.69 is whether any special treatment requirements are included in the new 50.44 rule that are based on bounding generic considerations that thorough the 50.69 process may be shown to be low safety significant to certain individual licensees.

Applicable NOPR Federal Register Locations:

Federal Register Page 26526 Column 3 Section III.4.9.3

"If portions of § 50.44 that were identified as special treatment requirements are retained, and/or relocated to other rules (and they are not necessary for RISC–3 SSCs), then there may be a need to reference these rules within § 50.69(b)(1) when § 50.69 is issued as a final rule."

C.16. DEFINITION OF SYSTEM FOR 50.69

Clarification should be provided regarding the definition of a "system" for the purposes of implementing the proposed 50.69 rule. In practice, the definition of a system varies from plant to plant in ways that may result in confusion regarding the intent of this section of the regulation. Several examples are:

- The use of tag numbers (unique component identifiers) to identify SSCs belonging to a common system can result in a different definition of the system boundaries compared to that used in the design basis documentation or the Maintenance Rule. For example, the feedwater and auxiliary feedwater

system may use a common component identifier, but are two separate systems from a safety perspective.

- The boundaries between mechanical and electrical components are not consistent from plant to plant. Using an MOV as an example, the boundary between the electrical "system" and "mechanical" system may be the electrical connections to the MOV at one plant and the motor control center at another plant.

It is recommended that each licensee be permitted the flexibility to define the "system" boundaries in a clear and consistent method for each plant.

Applicable NOPR Federal Register Locations:

Federal Register Page 26531	Column 1	Last Full Paragraph

"The Commission has determined that selective implementation on a system basis should be allowed, but not for components within a system."

Federal Register Page 26539	Column 3	Section V.4.5

"However, the implementation, including the categorization process, must address entire systems or structures; not selected components within a system or structure."

Federal Register Page 26540	Column 23	50.69(c)(1)(v)

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

D. EDITORIAL COMMENTS

D.1. PILOT PLANT SCOPE

The pilot plant activities had the objective of exercising the proposed NEI *and ASME categorization* guidance. All of the pilot activities only exercised the categorization process in the NEI guidance, as opposed to the entire NEI "Implementation" guidance. Additionally, the WOG pilot activities also exercised draft versions of the ASME Code Case for categorization of pressure boundary SSCs. The categorization of pressure boundary components using the ASME guidance provided significant feedback to the ASME that is reflected in the Code Case as approved by ASME.

Applicable NOPR Federal Register Locations:

Federal Register Page 26532	Column 1	Section IV.3.0

"To aid in the development of the proposed rule and associated implementation guidance, several plants volunteered to conduct pilot activities with the objective of exercising the proposed NEI implementation guidance and using the feedback and lessons-learned to improve both the implementation guidance and the governing regulatory framework."

D.2. TIMING OF CATEGORIZATION VS. NRC REVIEW

The licensee should not be required to wait until approval by NRC is received before proceeding with performing the categorization and treatment processes. In fact, the License Amendment Request required for implementation of 50.69 requires that some details of the categorization and treatment processes be provided for NRC review. The statement should more correctly read: "Upon approval of the categorization process, the licensee can begin implementation of treatment changes according to the approved categorization and treatment processes".

Applicable NOPR Federal Register Locations:

Federal Register Page 26533 Column 3 Bottom of Column

"Upon approval of the categorization process (and review of the supporting PRA), the licensee can begin implementation by performing categorization of SSCs and revising treatment requirements accordingly."

D.3. NRC REVIEW OF PRA

The reviewer guidance for PRA adequacy for use in categorization of SSCs for 50.69 is not provided in Section VI.2 of the Statement of Considerations as stated, but rather in Section VII.2.

Applicable NOPR Federal Register Locations:

Federal Register Page 26535 Column 1 Middle of Column

"The NRC has developed reviewer guidance applicable to these submittals and this is described below in Section VI.2."

and Federal Register Page 26535 Column 2 Middle of Column

"The NRC intends to use review guidance (discussed in more detail in Section VI) for this purpose."

and Federal Register Page 26536 Column 1 First Full Paragraph

"As discussed in Section VI, NRC has developed review guidelines for considering the sufficiency of a PRA that was subjected to the NEI peer review process, as it would be used in implementation of § 50.69."

D.4. ASME CODE CASE NUMBERS

The correct numbers of the ASME Code Cases are N-660 and N-662 for SSC categorization and treatment, respectively.

Applicable NOPR Federal Register Locations:

Federal Register Page 26541 Column 2 First Full Paragraph

"Further, the Commission expects that related standards (such as ASME Code Cases N–658 and N–660 on SSC categorization and treatment for purposes of repair and replacement) be used in conjunction with each other as intended by the accredited standards writing body."

E. SPECIFIC AREAS OF REQUESTED INPUT

E.1. NRC QUESTION ON CONSENSUS STANDARDS

Although alternate rule language on consensus standards is provided in the NOPR, we do not believe is necessary. If it should be adopted for whatever reason, then the terminology "meet voluntary consensus standard that is generally accepted in industrial practice applicable to replacement" needs significant clarification. As an example, there are voluntary consensus standards developed by ASME and API and some of the requirements from those standards are not consistent in areas of overlapping coverage. The intent of "use of applicable voluntary consensus standards" needs to be clarified, but not the extent as to foreclose the option of using voluntary consensus standards that may be developed in the future.

Applicable NOPR Federal Register Locations:

Federal Register Page 26546 Column 2 First Partial Paragraph

"Replacements for ASME Class 2 and Class 3 SSCs or parts must meet either: (1) The requirements of the ASME Boiler & Pressure Vessel (BPV) Code; or (2) the technical and administrative requirements, in their entirety, of a voluntary consensus standard that is generally accepted in industrial practice applicable to replacement."

E.2. ALTERNATE DEFINITIONS FOR BASIC COMPONENT

We agree with the Commission in rejecting the two alternatives. Both of the alternatives are less desirable from a practical implementation perspective than the primary method adopted by the NRC.

Applicable NOPR Federal Register Locations:

Federal Register Page 26522 Column 1 Last Full Paragraph

"The Commission does not believe that the part 21 definition of ''basic component'' may easily be read as simultaneously permitting both a deterministic concept of basic component and risk-informed concept, inasmuch as the part 21 definition was drawn from, and was intended to be consistent with the definition of ''safety-related SSC'' in § 50.2. .... The Commission rejected the second approach of adopting an alternative definition of ''basic component,'' because a change to the definition in § 21.3 could be misunderstood as a change to the reporting requirements for licensees who choose not to comply with § 50.69."

E.3. INSPECTION AND ENFORCEMENT GUIDANCE

We believe that it is appropriate to develop guidance and training for NRC inspectors who would be auditing licensees 50.69 programs to assure consistency in the Reactor Oversight Process in the manner intended by the new rule.

Federal Register Page 26529 Column 2 First Full Paragraph

"In its final decision on this rulemaking, the Commission proposes to document its conclusions as to whether new or revised inspection or enforcement guidance is necessary."

E.4. PUBLIC COMMENT ON ADDITIONAL RULE REQUIERMENTS VS. GUIDANCE

The current rule language, when coupled with the Regulatory Guide requirements for acceptable methodologies, should provide sufficient assurance to the NRC that the categorization process used by a licensee is robust.

Applicable NOPR Federal Register Locations:

Federal Register Page 26516 Column 1 1st Full Paragraph

"Public comment is requested on the merits of placing the additional detail shown in the guidance and discussed in Section V.4 of the Statement of Considerations (SOC) in the rule."

E.5. NRC QUESTION ON TREATMENT

We believe that the level of specification of the treatment requirements in the current rule language is adequate to assure that the SSCs categorized under an acceptable 50.69 process will perform their design basis functions. Several considerations, taken in aggregate, provide reasonable confidence that plant safety is not compromised by reduced treatment of low safety significant SSCs:

- the design basis functions for all SSCs will be maintained,
- operational considerations are considered by the Integrated Decision-Making Panel,
- defense in depth and safety margins are maintained, and
- risk assessment considerations provide assurance that there is a negligible change in risk

The robustness of the categorization process is used to assure that defense in depth, safety margins and risk are properly considered. There is no additional safety assurance provided by detailed requirements on treatment considerations for low safety significant SSCs. Therefore, we conclude that it is only appropriate to include the high level treatment requirements presented the NOPR. The licensee's submittal for NRC review and approval will include the attributes of the high level treatment to be applied to low safety significant SSCs, per the 50.69 rule language. Therefore, the SOC should be significantly revised to delete detailed expectations and requirements that do not directly support an explanation of the intent for the rule language.

Applicable NOPR Federal Register Locations:

Federal Register Page 26546 Columns 1&2 Section VI.1.0

"The Commission is requesting comment as to whether any of these requirements (or other requirements) are necessary to provide reasonable confidence of SSC functionality commensurate with the safety significance of the RISC–3 SSC, *i.e.*, whether the requirements on categorization are sufficiently robust that the level of detail contained in the proposed rule on treatment is appropriate."

E.6. NRC QUESTION ON PRA SCOPE

We believe that, for this application, the current requirements of the 50.69 rule are adequate to ensure that the categorization is robust and provides adequate assurance that reduced treatment applied to low safety significant SSCs will not adversely impact the plant risk. This conclusion is based on several observations and insights:

- NRC review and approval of the PRA is not necessary to assure the adequacy of the categorization process. The PRA quality requirements of DG-1122 that includes peer reviews, self assessments of the changes made to address significant peer review findings, and the additional requirements of the ASME PRA Standard (including NRC additions and exceptions) provide a robust basis for concluding that the internal events PRA is of adequate quality for categorization under 50.69. In addition, the NRC has interacted with each licensee regarding key aspects of their PRA during the individual plant SPAR model benchmarking exercises. This should provide an adequate level of review to preclude significant issues form being identified during a more in-depth NRC review of the licensee's entire PRA.
- The use of the qualitative risk assessments for fire (e.g., the FIVE methodology) and seismic (e.g., the Seismic Margins Analysis) would result in more SSCs being categorized as risk significant, compared to a quantitative fire or seismic PRA. For example, in the case of the Surry plant, the FIVE methodology was used to screen the fire scenarios that were modeled in the quantitative fire PRA. The number of risk significant SSCs from the quantitative fire PRA was significantly less than that from the FIVE methodology. An independent qualitative assessment of the fire PRA results for Surry also showed that no additional risk important SSCs could be expected that were not identified by the FIVE methodology. Therefore, there is assurance that the use of the qualitative risk assessments provide a bounding process for identifying potentially low risk significant SSCs.
- With respect to low power, transition and shutdown modes, the qualitative assessment methodologies in use by licensees are expected to result in more SSCs being categorized as safety significant compared to a quantitative shutdown PRA. The reason for this expectation is the extensive use of the shutdown methodologies by the licensee prior to (outage planning) and during (emergent issues) each plant outage. The experience gained by the licensee plant staff (who will be represented on the IDP) staff in understanding the risk significance of SSCs during a wide range of different plant outage

configurations provides a bounding basis for the 50.69 categorization process. Thus, the use of qualitative shutdown models is expected to provide a robust basis for determination of SSC risk importance during shutdown conditions.

- The PRA is only one input to the categorization process. Defense in depth and safety margin considerations are another important element of the categorization process. In addition, the requirement for an Integrated Decision-Making Panel to consider the any proposed categorization brings extensive wisdom and knowledge related to operations, maintenance and design of the candidate low safety significant SSCs. The entire categorization process assures that important considerations are not overlooked.

Applicable NOPR Federal Register Locations:

Federal Register Page 26546 Column 3 Section VI.2.1

"The Commission is seeking comment as to whether the NRC should amend the requirements in § 50.69(c) to require a level 2 internal and external initiating events, all-mode, peer-reviewed PRA that must be submitted to, and reviewed by, the NRC."

E.7. NRC QUESTION ON TREATMENT

We believe that the level of NRC review and approval of treatment processes specified in the current proposed rule language is adequate to assure that the SSCs will be capable of reliably performing their design basis functions. We believe that additional requirements or interactions are not a cost-effective use of the Commission's or the licensee's staff in terms or improving safety. This belief is rooted in several elements of the currently proposed rule:

- The NRC staff resources for review and endorsement of the generic categorization processes developed by NEI (NEI-00-04) and ASME (Code Case N-660) provides a high level of confidence that the categorization process will be robust. Subsequent NRC review and approval of the licensee's implementation of the endorsed processes provides a high level of confidence that the SSC categorization developed by the licensee is robust.
- Given that the categorization process is robust and that the sensitivity of degraded reliability for SSCs as a result of reduced treatment has been assessed and found to be acceptable in terms of increases in risk, a review of the detailed treatment processes to be applied to low safety significant SSCs would not add a incremental level of safety.
- NRC review of the details of treatment processes proposed by the licensee would likely be resource intensive because each review would involve a significant learning process to understand each licensees implementation of treatment processes for both safety related and non-safety related SSCs. Significant plant specific variations in the previously approved implementation methods are one of the primary reasons that the NRC staff resource requirements would remain high on subsequent reviews of the treatment processes.

Applicable NOPR Federal Register Locations:

Federal Register Page 26546 Column 3 Section VI.2.2

"For treatment requirements, the proposed rule sets forth high-level requirements, and does not require NRC review and approval of specific processes a licensee would implement to meet these requirements. Another way to structure the rule would be to require NRC review and approval of the licensee's proposed treatment program for RISC–3 SSCs. The Commission is interested in any benefits of this approach as well as any implications for this rulemaking and its associated guidance."

E.8. NRC QUESTION ON INSPECTION AND ENFORCEMENT

We believe that additional inspection and enforcement guidance is desirable in order to accommodate the significant changes in industry practices permitted by this rule. Specifically, licensees presently procure and install SSCs according to established procedures and processes. With the advent of RISC-3 relaxed treatment requirements and the attendant documentation and assessment requirements, a new set of procedures and processes will be developed and maintained as governed by the requirements of 50.69. This presents a new NRC inspection regime that will require new inspection guidance. Also, in the enforcement area, new guidance is required to enable a fair assessment of the potential risks presented by non-compliance findings. We encourage the Commission to develop the required inspection and enforcement guidance in conjunction with the development of the regulatory guidance for review and approval of the implementation of the 50.69 for systems and components.

Applicable NOPR Federal Register Locations:

Federal Register Page 26546 Column 3 Section VI.2.3

"Public comment is requested on whether or not changes are needed in our inspection and enforcement programs to enable NRC to exercise the appropriate degree of regulatory oversight of these aspects of the facility operation."

E.9. NRC QUESTION ON OPERATIONAL EXPERIENCE

We believe that a substantial body of operational experience for SSCs procured and installed to requirements similar to those that would be used for RISC-3 SSCs is currently available, but not summarized in a convenient format for the purposes of assessing uncertainty of the impact of relaxed treatment on SSC reliability. In particular, a large number of "balance-of-plant" SSCs are currently procured and installed according to a set of requirements that ensure their reliability. The overall purpose of these balance-of-plant SSC procurement and installation requirements is the need to keep reliability high so that the plant capacity remains as close to 100% as possible. This goal is developed by nuclear utilities for economic reasons – utility revenues are tied to the ability to keep the plant on-line and produce safe electric power. As a result of these balance-of-plant procurement and installation requirements, the plant capacity factors over the past decade have improved to record levels, in the 90% range. If the balance-of-plant components, whose procurement and installation are not regulated by the Commission, were not highly reliable, then such high capacity factors would not be achievable. This intuitively provides a level of confidence that SSCs procured and installed to nuclear balance-of-plant requirements will remain highly reliable and allow them to perform their design basis functions if required. Collection and assessment of data on the reliability of nuclear balance-of-plant SSCs would provide a quantitative measure of the intuitive level of confidence.

Applicable NOPR Federal Register Locations:

Federal Register Page 26547 Column 1 Section VI.2.4

"The Commission is interested in the role that relevant operational experience could play in reducing the uncertainty associated with the effects of treatment on performance and specifically seeks public comment as to what information might be available and how it could be used to support implementation of this rulemaking."

F DRAFT REGULATORY GUIDE DG-1121

F.1 REFERENCE TO NEI-00-04

The industry is in the process of revising Draft Revision C to NEI-00-04 to respond to the Staff comments (that are included as an Attachment to Draft Regulatory Guide DG-1121) as well as the results of discussions between the industry and the Staff at public meetings hosted by NRC at their Offices in White Flint. We would encourage the Staff to continue to provide venues for the industry and Staff to discuss any remaining issues with the industry guidance in NEI-00-04 with the aim of the Commission's endorsement of NEI-00-04 with minimal of clarifications, enhancements, and conditions.

Applicable Draft Regulatory Guide Locations:

Regulatory Guide Page 4 Section C.1

"Draft Revision C of NEI 00-04, "10 CFR 50.69 SSC Categorization Guidance," dated June 28, 2002 (ADAMS Accession No. ML021910534), provides an approach that is acceptable to the NRC staff for meeting the categorization requirements in the proposed 10 CFR 50.69 (68 FR 26511), subject to the following clarifications, enhancements, and conditions. [To be determined upon resolution of the issues discussed in the attachment, either by appropriate revisions to NEI 00-04 or by inclusion of staff positions in the Regulatory Guide]"

F.2 Required PRA Scope

The discussion of the required PRA scope for use in the proposed 50.69 categorization process should be revised to be consistent with the statements appearing in the Statement of Considerations and rule language of the proposed 50.69 in order to clarify the minimum required scope of the PRA for this application.

Specifically, the Statement of Considerations, at Federal Register page 26535, Column 3 Section V.4.1.1 describes the minimum PRA scope for 50.69 as "internal events occurring at full power operations". Additionally, at Federal Register pate 26537, bottom of second column, the use of risk assessment and risk management methodologies (e.g., fire and external events screening methodologies, seismic margins analyses, etc.) are described as acceptable methods to obtain insights for the categorization process for initiating events and plant operating models not modeled in the PRA. Further discussion of PRA scope to support implementation of the proposed 50.69 rule are provided in Section E.6 of this attachment.

Applicable Draft Regulatory Guide Locations:

Regulatory Guide Page 4 Section C.5

To meet the requirements of the proposed 10 CFR 50.69 for categorization of SSCs, licensees must use risk evaluations and insights that cover the full spectrum of potential events (i.e., internal and external initiating events) and the range of plant operating modes (i.e., full power, low power, and shutdown operations). The NRC staff believes that current state-of-the-art PRA methods are available to quantitatively address the full spectrum of potential events and the full range of plant operating modes for this type of application, and thus it is desirable for licensees to use such broad-scope PRAs. However, Draft Revision C of NEI 00-04 allows the use of non-PRA type evaluations (e.g., FIVE, seismic margins analysis, NUMARC 91-06), when PRAs have not been performed. It should be recognized that the degree of relief that can be expected will be commensurate with the assurance provided by the evaluation.