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WCAP-16308-NP-A
Revision 0

August 2009

Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station

Westinghouse

WCAP-16308-NP-A
Revision 0

Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station

Robert J. Lutz, Jr. *
Risk Applications and Methods I

August 2009

Approved: Melissa A. Lucci*, Manager
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Prepared for
Pressurized Water Reactor Owners Group

Developed under PWROG Program
PA-SEE-0027

*Electronically approved records are authenticated in the electronic document management system.

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March 26, 2009

Mr. Biff Bradley, Director
Risk Assessment
Nuclear Energy Institute
Suite 400
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Washington, DC 20006-3708

SUBJECT: FINAL SAFETY EVALUATION FOR NUCLEAR ENERGY INSTITUTE (NEI) TOPICAL REPORT (TR) WCAP-16308-NP, REVISION 0, "PRESSURIZED WATER REACTOR OWNERS GROUP [(PWROG)] 10 CFR [TITLE 10 OF THE CODE OF FEDERAL REGULATIONS] 50.69 PILOT PROGRAM – CATEGORIZATION PROCESS – WOLF CREEK GENERATING STATION" (TAC NO. MD4229)

Dear Mr. Bradley:

By letter dated September 25, 2006, as supplemented by letters dated October 22, 2007, and July 15, 2008, the NEI submitted Topical Report (TR) WCAP-16308-NP, Revision 0, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process- Wolf Creek Generating Station," to the U.S. Nuclear Regulatory Commission (NRC) staff for review on behalf of the PWROG. By letter dated September 16, 2008, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-16308-NP was provided for your review and comments. By letter dated October 14, 2008, NEI commented on the draft SE. The NRC staff's disposition of NEI's comments on the draft SE are discussed in Attachment 2 to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-16308-NP is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

The categorization process described in TR WCAP-16308-NP and evaluated in the enclosed final SE supports license amendment applications by licensees voluntarily choosing to implement 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." As described in the statement of considerations for 10 CFR 50.69, the treatment processes for structures, systems, and components (SSC) are not subject to NRC review and approval as part of the license amendment process, but rather will be subject to sample inspections at nuclear power plants implementing 10 CFR 50.69. Therefore, the SE does not approve nor endorse any specific treatment process. The NRC recognizes the need for an effective, stable and predictable inspection process regarding treatment requirements and looks forward to a continuing dialogue with NEI to ensure these goals are achieved.

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Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards. In accordance with the guidance provided on the NRC website, we request that NEI publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed final SE after the title page. In addition, the accepted version should be modified to incorporate the changes identified in Table 1 of the enclosed SE. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, NEI and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 689

Enclosure: Final SE

cc w/encl: See next page

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March 26, 2009

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/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16308-NP, REVISION 0

"PRESSURIZED WATER REACTOR OWNERS GROUP 10 CFR 50.69 PILOT PROGRAM – CATEGORIZATION PROCESS - WOLF CREEK GENERATING STATION"

NUCLEAR ENERGY INSTITUTE

PROJECT NO. 689

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 25, 2006 (Reference 1), as supplemented by letters dated October 22, 2007 (Reference 2), and July 15, 2008 (Reference 3), the Nuclear Energy Institute (NEI) submitted the Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 [Title 10 of the *Code of Federal* Regulation] Pilot Program – Categorization Process - Wolf Creek Generating Station," for U.S. Nuclear Regulatory Commission (NRC) staff review.

Reference 1 states that the primary objective of the submittal is to provide, for NRC review, a demonstration of a method for categorizing systems, structures, and components (SSCs) based on the safety significance of the pressure retaining function they perform (passive categorization). The TR refers to a "pilot" application of the proposed passive categorization methodology to two systems at the Wolf Creek Generating Station (WCGS). However, WCGS did not submit a request for licensing actions and no documentation on this pilot application was submitted. Therefore, as requested as the primary objective in Reference 1, the NRC staff has only reviewed the proposed passive categorization methodology described in the TR. This SE provides conclusions, findings, or endorsements of issues, methods, or results described in the TR for the proposed alternative method for passive categorization.

TR WCAP-16308-NP also provided a discussion of monitoring of Risk-Informed Safety Class (RISC)-1 and RISC-2 SSCs (provided in Section 7.2 of TR WCAP-16308-NP), monitoring of RISC-3 SSCs (provided in Section 7.3 of TR WCAP-16308-NP), and discussion of treatment of RISC-3 SSCs (provided in Section 8 of WCAP-16308-NP). Although it was not the primary objective of this TR, the NEI requested NRC feedback on the discussion located in Sections 7.2, 7.3, and 8 of TR WCAP 16308-NP. Therefore, Sections 3.4, 3.5, and 3.6 of this NRC staff safety evaluation (SE), provide the NRC staff's comments with respect to Sections 7.2, 7.3, and 8 of TR WCAP-16308-NP.

ENCLOSURE

2.0 REGULATORY EVALUATION

On November 22, 2004, the Commission adopted new Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," of 10 CFR on risk-informed categorization and treatment of SSCs for nuclear power plants (69 FR 68008). This new section permits power reactor licensees and license applicants to implement an alternative regulatory framework with respect to "special treatment," where special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that SSCs perform their design basis functions. Implementation of 10 CFR 50.69 requires that licensees first categorize safety-related and non-safety SSCs according to their safety significance. SSCs are classified into high-safety-significant (HSS) and low-safety significant (LSS) SSCs. Special treatment requirements for the LSS SSCs may be modified from those treatments otherwise required by the regulations as permitted by the rule.

In May of 2006, the NRC staff issued Regulatory Guide (RG) 1.201, "Guidelines For Categorizing Structures, Systems, And Components In Nuclear Power Plants According To Their Safety Significance, For Trial Use," Revision 1 (Reference 4). RG 1.201 describes a method that the NRC staff considers acceptable for use in complying with the Commission's requirements in 10 CFR 50.69 with respect to the categorization of SSCs that are considered in risk-informing special treatment requirements. RG 1.201 endorses a categorization method, with conditions, described in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, July 2005 (Reference 5).

NEI 00-04 describes, in detail, a methodology to categorize SSCs based on the active functions they perform (e.g., opening and closing of valves). Section 4.0 and Section 5.1 of NEI 00-04 references the American Society of Mechanical Engineers (ASME) Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities" (Reference 6), as an acceptable approach to categorize SSCs based on their passive functions. RG 1.201 clarifies that the version of ASME Code Case N-660 that is acceptable to the NRC staff for use in this application is the version identified in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (Reference 7).

TR WCAP-16308-NP proposes modifications to the methodology described in Code Case N-660 which results in an alternative method for passive categorization. The NRC staff evaluated the acceptability of this alternative method based on consistency with the requirements in 10 CFR 50.69, with the guidance endorsed in RG 1.201, and with the generic risk-informed decisionmaking guidelines established in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 8).

2.1 Monitoring of RISC-1 and RISC-2 SSCs

The regulation at 10 CFR 50.69(e)(1) requires that licensees shall review changes to the plant, operational practices, applicable plant and industry operational experience and, as appropriate, update the probabilistic risk assessment (PRA) and SSC categorization and treatment processes. Specifically for RISC-1 and RISC-2 SSCs, 10 CFR 50.69(e)(2) requires that licensees shall monitor the performance of these SSCs and make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid. In addition, RISC-1 SSCs are classified as safety-related and, therefore, are subject to the inservice inspection (ISI) and inservice testing (IST) requirements in

10 CFR 50.55a, "Codes and Standards," and the quality assurance requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," including Criterion XVI, "Corrective Action."

2.2 Monitoring of RISC-3 SSCs

The regulation at 10 CFR 50.69(e)(1) requires that licensees shall review changes to the plant, operational practices, applicable plant and industry operational experience and, as appropriate, update the PRA and SSC categorization and treatment processes. Specifically for RISC-3 SSCs, 10 CFR 50.69(e)(3) requires that licensees shall consider data collected in 10 CFR 50.69(d)(2)(i) to determine if there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations to satisfy 10 CFR 50.69(c)(1)(iv). The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization and results are maintained valid.

Under 10 CFR 50.69(d)(2)(i), licensees are required to conduct periodic inspections and tests to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions. In addition, 10 CFR 50.69(d)(2)(ii) requires that conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis conditions be corrected in a timely manner and, that for significant conditions adverse to quality, measures be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

Furthermore, 10 CFR 50.69(c)(1)(iv) requires that for RISC-3 SSCs, the categorization process must include evaluations that provide reasonable confidence that sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment are small.

2.3 Application of RISC-3 Treatment Requirements

The regulation at 10 CFR 50.69(d)(2) requires that licensees or applicants shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC-3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action shall be provided for RISC-3 SSCs.

3.0 TECHNICAL EVALUATION

The TR proposes a number of changes to ASME Code Case N-660. All of the proposed changes are described in Reference 3, Table A-2. Making the changes in Table A-2 to ASME Code Case N-660 will result in a description of the proposed alternative method. Table A-2 also briefly describes and explains the reason for each of the proposed changes.

Table 1 attached to this SE includes all the proposed changes from Table A-2 although, in some cases, several entries in Table A-2 have been combined into a single entry in Table 1.

Table 1 provides the NRC staff position on each proposed change, including NRC revisions as applicable. These positions are:

• **No objection**. The NRC staff has no objection to the requirement.
• **Objection requiring qualification**. The NRC staff has a technical concern with the requirement and has provided a qualification to resolve the concern.

As stated in Table 1, the NRC staff accepts most proposed changes with "No objection." However, the NRC staff accepts the remaining proposed changes with "Objection requiring qualification." The qualifications are provided as revisions to the proposed text where additions are identified by bolded text, deletions by struck out text. Making the changes in Table 1 to the methodology described in ASME Code Case N-660 will result in a description of the alternative method that can be endorsed in this SE.

Many of the proposed changes improve the consistency between the categorization methodology in NEI 00-04 and methodology proposed in the TR. For example, the TR reorganized the sections dealing with the treatment of various qualitative considerations which resulted in numerous individual changes. The NRC staff accepted most of these changes with no objection. In several proposed changes, the NRC staff has added a qualification to improve the consistency between the two categorization methods. For example, the NRC staff added text from NEI 00-04 into Section I-3.1.2. Section I-3.1.2 directs that risk information from all initiating events be included in the categorization. The additional text clarifies one acceptable method to capture risk importance from initiating events that are not modeled in a PRA.

In addition to conforming changes, the TR also proposed several substantive changes to the methodology in Code Case N-660. Substantive changes include proposals to delete some qualitative considerations, expand credit for operator actions, and permit consequence determination based on small break sizes. These proposed changes are discussed below.

3.1 Deletion of Some Qualitative Considerations

Consistent with NEI 00-04, ASME Code Case N-660 provides a series of questions that need to be considered by the licensee's personnel when assigning an SSC into the HSS or LSS category. The response to these questions support the systematic determination on whether SSCs that are not assigned HSS by the quantitative PRA results, should be assigned HSS based on qualitative considerations, including defense-in-depth and safety margins considerations.

The question in ASME Code Case N-660 Section I-3.1.3(a)(2) was deleted. The response to this question would require that all piping as defined in ASME Code Case N-660 Section 1200(b) be assigned to the HSS category. Section 1200(b) already assigns this piping to HSS and the NRC staff concurs that this question is redundant and may be deleted.

The question in ASME Code Case N-660 Section I-3.1.3(b)(1) was deleted. The TR states that the response to this question would require that all piping in every system that supports the retention of fission products during severe accidents be assigned to the HSS category. The NRC staff agrees that the ASME Code Case N-660 guidance is conservative because it would place whole systems into the HSS category based on small, and perhaps very small, parts of the system acting as a barrier to fission product release.
All of the effects of piping rupture, including the potential to cause or permit a release during a severe accident, are addressed as part of the passive categorization process. The conditional large early release probability (CLERP) guidelines should identify those piping parts in a system whose failure contributes significantly to fission product release as HSS segments. The NRC

staff concurs that the question in ASME Code Case N-660 Section I-3.1.3(b)(1) may be deleted because it is excessively conservative, and a question with the excessive conservatism removed is not expected to identify any piping as HSS piping that would not otherwise be identified.

The question in ASME Code Case N-660 Section I-3.1.3(a)(1) was deleted. ASME Code Case N-660 and the alternative method, as endorsed in this SE, categorize the passive functions of SSCs based on the quantitative PRA metrics conditional core damage probability (CCDP) and CLERP[1]. This question introduced PRA metrics based on the potential for pipe rupture events to increase the frequency of non-pipe rupture initiating events. However, all the effects of a pipe rupture, including all initiating events it causes, are already addressed as part of the passive categorization process. The NRC staff concurs that this question may be deleted because categorization based on PRA results is adequately addressed in Section I-3.1, and these additional quantitative metrics are not expected to identify any HSS piping that would not otherwise be identified.

3.2 Expanded Credit for Operator Actions

The TR WCAP-16308-NP proposed to add guidance that would direct the expert panel to credit possible operator actions in the qualitative responses to the questions in ASME Code Case N-660 Sections I-3.1.3(a)(5) and I-3.1.3(b)(3) (modified and moved to questions I-3.2.2(b)(3) and I-3.2.2(b)(6) respectively in the TR). This proposal would allow the licensee's expert panel to qualitatively decide that the undesired consequence postulated in the question could be avoided because, and only because, the operators took action. The TR argues that its proposal only permits credit if a procedure directs the operators' response. However, symptom based procedures often direct the operators, in general, to develop and attempt mitigative actions and, therefore, any conceivable mitigative actions would arguably satisfy the criterion.

The NRC staff does not accept the proposal to credit operator action in these two questions. The direct consequences of pipe failures addressed in these questions are (a) inability to reach and maintain shutdown and (b) large radioactive material release. Therefore, the conditional core damage probability and large early release frequency for these pipe breaks, without operator action, is 1.0. In order to reduce the conditional risk from these pipe breaks to below the HSS guideline values, the probability of the operators failing to accomplish the task should be 10^{-4} or less, a very small human error probability. The NRC staff does not believe that a qualitative judgment by the expert panel about operator actions following such uncommon events such as pipe ruptures can routinely support this determination.

The NRC staff concludes that the expert panel's qualitative judgment is not sufficient to identify operator actions that have very low failure probabilities but also generally recognizes that risk-informed evaluations should reasonably credit operator actions. Therefore, the NRC staff further modified the question proposed in I-3.2.2(b)(3) of the TR to be consistent with both these positions.

[1] Code Case N-660 and the proposed alternative method permit the use of tables instead of the quantitative guidelines directly, but the entries in the tables were derived from the quantitative CCDP and CLERP guidelines values.

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The question is modified from:

- Even when taking credit for plant features and operator actions, failure of the piping segment will not prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions.

To:

- ~~Even when taking credit for plant features and operator actions~~, failure of the piping segment will not prevent ~~or adversely affect~~ the plant~~'s capability to reach or maintain~~ reaching or maintaining safe shutdown conditions.

A piping failure which only "adversely affects" a plants shutdown capabilities would have a CCDP and CLERP less than 1.0 and the operator actions to mitigate only degraded, and not failed, shutdown capability need not be so highly reliable in order to be consistent with the quantitative guidelines. Degraded safety-significant functions are addressed in the TR question I-3.2.2(b)(1) which does permit qualitative consideration of operator actions.

When extensive engineering judgment is required in an evaluation, the NRC staff may require that the evaluation be submitted to the NRC to allow prior staff review and approval. Extensive engineering judgment is required to properly characterize very reliable operator actions following such unusual events as piping ruptures. If a licensee wants to credit operator actions in these particular circumstances described in ASME Code Case N-660 Sections I-3.1.3(a)(5) and I-3.1.3(b)(3), it may take this step and report its evaluation to the NRC as a deviation from the method described in the approved methodology. The NRC would then review the particular operator action, associated circumstances and the documentation supporting the conclusion and agree or disagree with the licensee expert panel's conclusion that the documentation demonstrates that the likelihood of success of the operator action is indeed very high.

3.3 Consequence Evaluation Based on Small Break Size

Section I-3-1.1(a) in ASME Code Case N-660 required that the consequence analysis be performed assuming a large pressure boundary failure unless one or more of the three criteria could be met. If any one of these criteria was met, a smaller break could be assumed when determining the affects of the pressure boundary failure. Smaller breaks tend to result in damage to fewer nearby SSCs and slower transients than larger breaks. Assessing the consequence for small instead of large breaks could result in assigning a lower safety significance to pressure boundary failures.

The first of the three criteria in ASME Code Case N-660 simply permits the consequences of a smaller leak to be used if more conservative than using a larger break. The second and third criteria, when met, provide confidence that a large break is very unlikely according to NRC endorsed methods regardless of how the piping in question is repaired or replaced.

The TR proposes to add a fourth criterion in I-3-1.1(a)(4):

{Alternatively, the consequence analysis can be performed assuming a smaller leak, when} a small break with a calculated leak rate at design basis conditions for a through-wall flaw with a length six times its depth can be used when certain design and operational considerations are satisfied:

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- *the pipe segment is not susceptible to any large break mechanisms or plant controls are in place to minimize the potential for occurrence of large break mechanisms,*
- *a large break mechanism is one that produces significant loadings above the normal loading on the system and specifically includes water hammer for which no mitigation is provided and internal deflagrations, but excludes seismic,*
- *the pipe segment is not part of a high energy system, and*
- *the pipe segment is greater than 4 inches in diameter.*

The NRC staff has evaluated the reasonableness of the specific criteria proposed by the NEI as supported by the Request for Additional Information (RAI) response in Reference 2.

1) Reference 2 cites insights taken from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," regarding the low likelihood of pipe ruptures. However, NUREG-1829 only applies to the reactor coolant pressure boundary (high pressure system) and can not be used as a basis to draw conclusions regarding the probability of failure for low pressure systems (e.g., service water systems). The reactor coolant pressure boundary is built and maintained to the highest quality standards. In addition, leak-before-break evaluations have been performed for numerous facilities in order to demonstrate a low probability of failure. The low pressure systems are not subject to the same quality standards as the reactor coolant pressure boundary.

2) Reference 2 also cites the use of earthquake experience in resolution of Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," as providing evidence of the capacity of the piping to withstand seismic loads. The NRC staff did not endorse the use of earthquake experience for piping in the resolution of GL 87-02. However, the existing earthquake experience data indicates that low pressure systems are more likely to fail during earthquakes than high pressure systems (see NUREG-1061, Volume 2 Addendum, Section 2.2.7).

3) Reference 2 also states that a pipe diameter of 4 inches was selected to coincide with the ASME definition of small bore piping. The NRC staff does not concur that this is the ASME definition of small bore piping.

4) Reference 2 states that the appropriate small break size for consideration in passive components is the calculated leak rate at normal conditions for a through-wall flaw with a length six times its depth. This discussion cites NUREG-1829 as part of the basis for this assumption. However, as discussed above, NUREG-1829 only applies to the reactor coolant pressure boundary and its evaluations and conclusions cannot be extrapolated to low pressure service water systems.

Implementation of 10 CFR 50.69 permits licensees to change the special treatment requirements applied to LSS SSCs. The failure frequency of piping is generally not well known. Instead of attempting to estimate the frequency of piping failure, passive categorization is based on the consequence of failure. Any piping segment with a CCDP or CLERP greater than 1E-4 and 1E-5 respectively, will be HSS. Therefore, at the (unlikely) limit where the failure likelihood of a LSS segment approaches 1.0, there is a known upper bound on the risk increase. If, however, the consequences of a small break are used instead of a large break, the CCDP or CLERP of a large break in an LSS segment could exceed the guideline values by an indeterminate magnitude. In ASME Code Case N-660, the NRC staff only accepted use of the consequences based on the small break only if the larger break is very unlikely based on the

results of analyses endorsed by the NRC staff.

The NRC staff relies on the limitation in the potential risk increase provided by categorization based solely on the consequences of a pipe break to satisfy the criterion in 10 CFR 50.69(c)(1)(iv) that any potential increase in risk is small. The guidelines proposed in the TR are not endorsed for use in the piping systems that will be categorized because they do not provide the necessary confidence that the large break is very unlikely. Therefore, the NRC staff concludes that the proposal to include additional guidelines permitting the use of smaller breaks is not acceptable.

3.4 Monitoring of RISC-1 and RISC-2 SSCs

TR WCAP-16308-NP states that the Wolf Creek Nuclear Operating Corporation (WCNOC), the licensee for WCGS, intends to comply with 10 CFR 50.69 without exception. RISC-1 and RISC-2 SSCs will be monitored in the same manner as they are presently monitored under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (the Maintenance Rule). As clarification, the NEI stated that monitoring will address all functional failures, not just maintenance preventable functional failures; and that to the extent that any RISC-1 or RISC-2 SSCs are not in the Maintenance Rule scope, appropriate monitoring requirements will be developed for those SSCs.

In addition to monitoring under 10 CFR 50.65, RISC-1 SSCs are subject to the regulatory requirements for safety-related equipment specified in 10 CFR Part 50. For example, SSCs within the scope of the ASME Boiler and Pressure Vessel Code and ASME Code for Operation and Maintenance of Nuclear Power Plants are required to meet the ISI and IST requirements specified in 10 CFR 50.55a. Among those requirements is the IST provision for periodically assessing the operational readiness of pumps and valves to perform their safety functions, and the ISI provisions that require a mandatory program of examinations, pressure testing, and inspections for determining component acceptability for continued service and to manage deterioration and aging effects, along with repair/replacement activity requirements. Further, Quality Assurance Criterion XVI, "Corrective Action," in 10 CFR Part 50, Appendix B, states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, Criterion XVI requires that the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

The NRC staff finds that monitoring of RISC-1 and RISC-2 SSCs as specified under 10 CFR 50.65, or through appropriate equivalent requirements for non-maintenance rule scoped items, together with the safety-related requirements for monitoring RISC-1 SSCs, provide an adequate means of monitoring these SSCs such that the results of this monitoring can be used to adjust the categorization or treatment processes so that the categorization and results are maintained valid. This conclusion is based on the safety-related requirements for RISC-1 SSCs and the requirements in 10 CFR 50.65 applicable to RISC-1 and RISC-2 SSCs that the performance or condition of SSCs be monitored in a manner sufficient to provide reasonable assurance that in-scope SSCs are capable of fulfilling their intended functions. Therefore, the NRC staff finds that the performance of monitoring, in accordance with safety-related requirements for RISC-1 SSCs, and the implementation of a monitoring program that satisfies 10 CFR 50.65, or an appropriate equivalent, for the purpose of RISC-1 and RISC-2 SSCs, provide an adequate means of satisfying the monitoring requirements of 10 CFR 50.69(e)(2).

3.5 Monitoring of RISC-3 SSCs

TR WCAP-16308-NP states that performance monitoring of RISC-3 SSCs will be established to provide assurance that potential increases in failure rates will be detected and addressed before reaching the rate assumed in the sensitivity study. Failures of RISC-3 SSCs will be identified and tracked in a corrective action program. Failure data will be periodically assessed to ensure the failure rate of RISC-3 SSCs has not unacceptably increased due to the changes in treatment and to validate that the rate of equipment failures has not increased by a factor greater than that used in the sensitivity studies. Component group failure data will also be reviewed to detect the occurrence of potential inter-system common cause failures and to allow timely corrective action if necessary.

The NRC staff notes that NRC Regulatory Guide (RG) 1.201, revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," describes an acceptable method for the categorization of SSCs based on safety significance; and that the endorsed Nuclear Energy Institute (NEI) guidance document NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," provides an acceptable method for adjusting categorization based on the performance monitoring of RISC-3 SSCs. RG 1.201 does not, however, endorse a particular treatment strategy for RISC-3 SSCs, including the specific inspection, testing and corrective action plans that would be used to ensure that degradation and common cause failure concerns will be addressed.

Although the TR discusses how failure data for RISC-3 SSCs will be used to satisfy the monitoring requirements of 10 CFR 50.69(e)(3) with respect to the adjustment of categorization and treatment based on unreliability values used in the sensitivity study, it does not contain a specific discussion of how monitoring for degradation of RISC-3 SSCs and corrective actions will be accomplished. In a response to an NRC staff RAI regarding corrective action for degradation of RISC-3 SSCs, the NEI stated that WCGS has not developed plant specific methods for corrective actions to address degradation of RISC-3 SSCs. For example, specific information on periodic inspections and tests that could be used to detect and correct degradation of RISC-3 SSCs was not provided. Although the staff concludes that Section 7.3 of TR WCAP-16308-NP describes a process for adjusting categorization and treatment, based on the results of monitoring RISC-3 SSCs and impacts on the unreliability values used in the sensitivity study, that satisfies 10 CFR 50.69(e)(3), there is insufficient information to determine if the monitoring and corrective actions for degradation of RISC-3 SSCs would provide timely correction of conditions that would prevent these SSCs from performing their safety functions, as required by 10 CFR 50.69(d)(2)(ii). In particular, licensees implementing 10 CFR 50.69 will need to address the cause of SSC degradation and preclude its repetition.

3.6 Application of RISC-3 Treatment Requirements

The NRC staff notes that 10 CFR 50.69(b)(2) does not require that a licensee voluntarily choosing to implement the rule submit their plan for treatment of SSCs to the NRC for review and approval. Although the primary objective of TR WCAP-16308-NP is to provide documentation of the categorization process used by WCGS to support implementation of 10 CFR 50.69, Section 8 of the TR also contains a brief description of the application of RISC-3 treatment requirements. Therefore, the staff is providing the following evaluation regarding treatment of RISC-3 SSCs as described in WCAP-16308-NP.

TR WCAP-16308-NP states that WCGS will develop and implement documented processes to control the design, procurement, inspection, and maintenance to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions. In its response to an NRC staff RAI, the NEI stated that WCGS has not developed plant specific methods for inspection, testing, and corrective actions for RISC-3 SSCs to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. In addition, the response stated that WCGS would apply commercial grade practices to the procurement, maintenance, and testing of RISC-3 SSCs.

The NRC staff finds that the general information provided in TR WCAP-16308-NP and in the RAI response does not provide a sufficient basis for determining that the regulatory requirements of 10 CFR 50.69(d)(2) would be satisfied.

The general reference by the NEI to the use of commercial grade practices in its response to the RAI on Section 8 of TR WCAP-16308-NP does not provide reasonable confidence in the functionality of RISC-3 SSCs, given the wide range of quality activities applied to these practices and their varying levels of effectiveness. For example, in the *Federal Register* notice (69 FR 68008, 68041) announcing issuance of 10 CFR 50.69, the NRC noted that some public comments on the proposed rule suggested that a reference to general industry practices would be sufficient to satisfy the requirements for treatment of RISC-3 SSCs. The NRC referred to NUREG/CR-6752, "A Comparative Analysis of Special Treatment Requirements for Systems, Structures, and Components (SSCs) of Nuclear Power Plants With Commercial Requirements of Non-Nuclear Power Plants," which found that significant variation exists in the application of industrial practices at nuclear power plants. The NRC stated that a simple reference to these practices does not provide a basis to satisfy the rule's requirements.

The regulation at 10 CFR 50.69(d)(2)(ii) requires that conditions that would prevent a RISC-3 SSC from performing its safety-related function under design-basis conditions must be corrected in a timely manner. Section 8 of WCAP-16308-NP refers to Section 7 of the TR for a discussion of the WCGS approach to RISC-3 corrective action. As described previously, the NRC staff finds that the TR and RAI responses rely primarily on monitoring of SSC failures and do not provide sufficient information to draw a conclusion that degradation of RISC-3 SSCs will be monitored and corrected in a manner that provides reasonable confidence that these SSCs would continue to perform their safety-related functions under design-basis conditions.

Therefore, licensees electing to implement 10 CFR 50.69 would be expected to implement a treatment program for RISC-3 SSCs that contains elements beyond a simple reference to commercial practices and monitoring of failure rates to provide reasonable confidence that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life.

4.0 LIMITATIONS AND CONDITIONS

1. This NRC staff safety evaluation (SE) only provides conclusions, findings, or endorsements for the proposed passive categorization methodology described in the TR. It does not provide conclusions, findings, or endorsements for issues, methods, or results outside the scope of the passive categorization methodology.

2. The alternative method proposed by the TR is described by modifying the method described in ASME Code Case N-660 according to the changes described in Reference 3, Table A-2. The NRC staff does not find this alternative method acceptable but, instead, would endorse the method described by modifying ASME Code Case N-660 according to the changes described in Table 1 of this SE. The TR should be modified in the approved (-A) version of the TR, to incorporate the changes identified in Table 1 of this SE. The NRC staff will not accept submittals referencing TR WCAP-16308-NP as an approved passive categorization methodology unless the method used in the submittal incorporates the changes identified in Table 1 as "Objection requiring qualification."

3. As described in Section 3.2 of the SE, the NRC staff does not accept the proposal to add guidance that would direct the expert panel to credit possible operator actions in the qualitative responses to the questions in ASME Code Case N-660 Sections I-3.1.3(a)(5) and I-3.1.3(b)(3). Operator actions would need to be very reliable in order to be appropriately relied on to reduce the consequences of these pipe ruptures below the guideline values. The NRC staff does not believe that a qualitative judgment by the expert panel about operator actions following such uncommon events such as pipe ruptures can routinely support a determination that the action is very reliable. Licensees that want to credit operator actions in the response to these questions should report their evaluation to the NRC as a deviation from the NRC-approved methodology.

4. As described in Section 3.3 of the SE, the NRC staff relies on the limitation in the potential risk increase provided by categorization based solely on the consequences of a pipe break to satisfy the criterion in 10 CFR 50.69(c)(1)(iv) that any potential increase in risk is small. The guidelines proposed in the TR are not endorsed for use in the piping systems that will be categorized because they do not provide the necessary confidence that the large break is very unlikely. Therefore, the NRC staff concludes that the proposal to include additional guidelines permitting the use of smaller breaks is not acceptable.

5. Licensees that implement 10 CFR 50.69 must develop and implement plant-specific programs to ensure that monitoring and corrective actions for degradation of RISC-3 SSCs will ensure the requirements of 10 CFR 50.69(d)(2)(ii) will be met. In particular, licensees implementing 10 CFR 50.69 will need to address the cause of SSC degradation and preclude its repetition.

6. Licensees that implement 10 CFR 50.69 must develop and implement plant-specific programs to ensure that treatment of RISC-3 SSCs is in accordance with 10 CFR 50.69(d)(2) and provides reasonable confidence that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions.

5.0 CONCLUSIONS

The NRC staff has found that only portions of the alternative methodology proposed in TR WCAP-16308-NP, Revision 0, are acceptable. The NRC staff has identified the specific items in the proposal that are not acceptable in this SE. Table 1 of this SE, identifies changes to the method proposed in WCAP-16308-NP that, if made, will result in an acceptable method to categorize SSCs as HSS or LSS based on the safety-significance of the passive functions they perform. Each change, and the NRC staff position on each change, is described in Table 1. A description of the approved methodology can be obtained by modifying the guidance in ASME Code Case N-660 as described in Table 1.

The NRC staff will not repeat its review of the matters described in WCAP-16308-NP, Revision 0 as modified by Table 1 in this SE, when the report appears as a reference in a request to amend a licensee's operating license to comply with the requirements of 10 CFR 50.69.

5.1 Monitoring of RISC-1 and RISC-2 SSCs

The NRC staff has reviewed the description of monitoring of RISC-1 and RISC-2 SSCs provided in Section 7.2 of TR WCAP-16308-NP. The NRC staff concludes that the TR has adequately addressed the monitoring of RISC-1 and RISC-2 SSCs for the WCGS 10 CFR 50.69 pilot program. Therefore, the NRC staff finds Section 7.2 of TR WCAP-16308-NP to be acceptable with respect to the monitoring of RISC-1 and RISC-2 SSCs together with the safety-related requirements for RISC-1 SSCs.

5.2 Monitoring of RISC-3 SSCs

The NRC staff has reviewed the discussion of monitoring of RISC-3 SSCs provided in Section 7.3 of TR WCAP-16308-NP. Based on its review, the NRC staff concludes that the TR describes an adequate means for adjusting the categorization or treatment process, based on the results of monitoring RISC-3 SSCs and the unreliability values used in the sensitivity study, in accordance with 10 CFR 50.69(e)(3) for the WCGS 10 CFR 50.69 pilot program. However, the NRC staff finds that Section 7.3 of TR WCAP-16308-NP does not provide sufficient information on monitoring and correction of degradation of RISC-3 SSCs to conclude that conditions that would lead to adverse changes in performance of RISC-3 SSCs will be monitored and corrected in accordance with 10 CFR 50.69(d)(2)(ii). Licensees that implement 10 CFR 50.69 must develop and implement plant-specific programs to ensure that monitoring and corrective actions for conditions that could prevent RISC-3 SSCs from performing their safety-related functions under design basis conditions will be implemented in a timely manner as required by 10 CFR 50.69(d)(2)(ii).

5.3 Application of RISC-3 Treatment Requirements

The NRC staff has reviewed the discussion of treatment of RISC-3 SSCs provided in Section 8 of WCAP-16308-NP. The NRC staff concludes that the TR has not adequately addressed the treatment of RISC-3 SSCs for the WCGS 10 CFR 50.69 pilot program. Specifically, the NRC staff finds that Section 8 of WCAP-16308-NP does not contain sufficient information on treatment of RISC-3 SSCs to provide reasonable confidence that RISC-3 SSCs will continue to perform their safety-related functions under design-basis conditions consistent with 10 CFR 50.69. Licensees that implement 10 CFR 50.69 must develop and implement plant-

specific programs to ensure that treatment of RISC-3 SSCs is in accordance with 10 CFR 50.69(d)(2) and provides reasonable confidence that RISC-3 SSCs will remain capable of performing their safety-related functions under design-basis conditions.

6.0 REFERENCES

1. B. Bradley (Nuclear Energy Institute) to U. S. Nuclear Regulatory Commission, Submittal of Topical Report on 10 CFR 50.69 Application, September 25, 2006 (ML062770345).

2. B. Bradley, (Nuclear Energy Institute) to U. S. Nuclear Regulatory Commission, Response to NRC Request for Additional Information Regarding WCAP-16308-NP, 10 CFR 50.69 Application, October 22, 2007 (ML080780403).

3. B. Bradley, (Nuclear Energy Institute) to U. S. Nuclear Regulatory Commission, "Revisions to WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process," July 15, 2008 (ML082200518).

4. RG 1.201, "Guidelines For Categorizing Structures, Systems, And Components In Nuclear Power Plants According To Their Safety Significance, For Trial Use," Revision 1 (ML061090627).

5. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, July 2005 (ML0529001630).

6. ASME Code Case, N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," July 2002.

7. RG 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (MI072070419).

8. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ML023240437).

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Attachments: 1. Table 1
2. Resolution of Comments

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Date: March 26, 2009

Table 1 – NRC Staff Position on Proposed Changes in ASME Code Case N-660 in TR WCAP-16308

{N-660, R0 Section} [WCAP-16308 Section]	**ASME Code Case N-660 Revision 0**	**Position**	**Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic**
{-1320} [-1320]	-1320 Required Disciplines Personnel with expertise in the following disciplines shall be included in the classification process. *(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.	No Objection	-1320 Required Disciplines (a) An Integrated Decisionmaking Panel (IDP) shall use the information and insights compiled in the initial categorization process and combine that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of functions/SSCs. (b) The designated as members of the IDP shall have joint expertise in the following fields: - Plant Operations (SRO qualified), - Design Engineering, - Safety analysis, - Systems Engineering, and - Probabilistic Risk Assessment. (c) Requirements for ensuing adequate expertise levels and training of IDP members in the categorization process shall be established. (d) To the extent possible, the classification of pressure retaining and support items in a system should be performed by the same IDP members as the categorization of active SSCs in that system.“

ATTACHMENT 1

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{-9000} [-9000]	high-safety-significant function – a function that has been determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events (e.g., evaluations performed to support the Maintenance Rule - 10 CFR 50.65). probabilistic risk assessment (PRA) – a qualitative and quantitative assessment of the risk associated with plant operation and maintenance... spatial effect – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, or flooding.	No objection	high-safety-significant function – a function that has been determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events (e.g., evaluations performed to support the Maintenance Rule - 10 CFR 50.65) or from other relevant information (e.g., defense in depth considerations) probabilistic risk assessment (PRA) – an assessment of the risk associated with plant operation and maintenance... spatial effect – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, loss of inventory due to draining of a tank, or flooding.
{ } [-9000]		No objection	Plant features – systems, structures, and components that can be used to prevent or mitigate an accident.
{I-1.0} [I-1.0]	Once categorized, the safety significance of piping of each piping segment is identified.	No objection	Once categorized, the safety significance of piping of each piping segment is identified. Figure I-1 illustrates the RISC methodology presented in the following sections. [Figure I-1]

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-2.0} [I-2.0]	The owner shall define the boundaries included in the scope of the RISC evaluation process.	Objection requiring qualification	The owner shall define the boundaries included in the scope of the RISC evaluation process **subject to the constraints in paragraph 50.59(c)(1)(v) that the categorization must be performed for entire systems**. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply."
{I-3.0} [I-3.0]	CONSEQUENCE ASSESSMENT	No objection	EVALUATION OF RISK INFORMED SAFETY CLASSIFICATIONS
{I-3.0} [I-3.0]	Piping segments can be grouped based on common conditional consequence...	No objection	All pressure retaining items, including supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common conditional consequence...

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.0} [I-3.0]	Additionally, information shall be collected for each piping segment that is not modeled in the PRA, but considered relevant to the classification (e.g., information regarding design basis accidents, shutdown risk, containment isolation, flooding, fires, seismic conditions).	Objection requiring qualification	Changed to read, "Additionally, information considered relevant to the classification shall be collected for each piping segment (e.g., information regarding design basis accidents, at-power risk, shutdown risk, containment isolation, flooding, fires, seismic conditions, etc.). **Consistent with 50.69(c)(1)(ii), the classification must address all initiating events and plant operating modes.** ~~This other relevant information is considered in conjunction with the Consequence Category to determine the Risk-Informed Safety Classification. The Consequence Category is Determined from the Consequence Evaluation.~~"
{I-3.1.1} [I-3.1.1]	Potential failure modes for each piping segment shall be identified...	No objection	Potential failure modes for each system or piping segment shall be identified...

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{ } [I-3.1.1(a)(4)]		Objection requiring qualification	*Entire proposed section should be deleted* ~~(4) a small break with a calculated leak rate at design basis conditions for a through wall flaw with a length six times its depth can be used when certain design and operational considerations are satisfied:~~ ~~- the pipe segment is not susceptible to any large break mechanisms or plant controls are in place to minimize the potential for occurrence of large break mechanisms,~~ ~~+ a large break mechanism is one that produces significant loadings above the normal loading on the system and specifically includes water hammer for which no mitigation is provided and internal deflagrations, but excludes seismic,~~ ~~- the pipe segment is not part of a high energy system,~~ ~~- the pipe segment is greater than 4 inches in diameter.~~
{-3.1.1(c)} [I-3.1.1(c)]	Indirect Effects. These include spatial interactions such as pipe whip, jet spray, and loss of inventory effects (e.g., draining of a tank).	No objection	Indirect Effects. A failure consequence affecting other systems or components, such as spatial effects.
{-3.1.1(d)} [I-3.1.1(d)]	Initiating Events. These are identified using a list of initiating events from any existing plant specific Probabilistic Risk Assessment (PRA) or Individual Plant Examination (IPE) and the Owner's Requirements.	No objection	Initiating Events. For systems or piping segments that are modeled either explicitly or implicitly in any existing plant-specific Probabilistic Risk Assessment (PRA), any applicable initiating event is identified using a list of initiating events from that PRA.
{-3.1.2} [I-3.1.2]	... (high, medium, low)...	No objection	... (high, medium, low, or none)...

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.2} [I-3.1.2]	... in accordance with (a) through (d) below.	Objection requiring qualification	... in accordance with (a) through (d) below. In assessing the appropriate consequence category, risk information for all initiating events, including fire and seismic, should be considered. **To capture the risk importance from initiating events for which no quantitative PRA is available, any piping segment supporting a safe shutdown pathway would be classified as HSS.** "
{I-3.1.2(a)(1)} [I-3.1.2(a)(1)]	The initiating event shall be placed in one of the categories in Table I-1.	No objection	The initiating event shall be placed in one of the Design Basis Event Categories in Table I-1.
{I-3.1.2(a)(1)} [I-3.1.2(a)(1)]	... updated final safety analysis report, PRA, or IPE shall be included.	No objection	... updated final safety analysis report or PRA shall be included
{I-3.1.2(b)(1)} [I-3.1.2(b)(1)]	Frequency of challenge that determines how often the mitigating function of the system is called upon. This corresponds to the frequency of initiating events that require the system operation.	No objection	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."
{I-3.1.2(b)(3)} [ ]	Exposure time shall be obtained from Technical Specification limits.	No objection	*Direction may be deleted because the same direction appears earlier in the paragraph.*
{I-3.1.2(b)(3)} [I-3.1.2(b)]	In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5.	No objection	In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5.
{I-3.1.2(d)} [I-3.1.2(d)]	The above evaluations determine failure importance relative to core damage.	No objection	The above evaluations determine failure importance relative to core damage or the plant's capability to reach or maintain safe shutdown conditions."

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.3} {I-3.1.3(b)} {I-3.2.2(b)} [I-3.2.2(b)]	If any of the conditions in (a) or (b) below are true, the piping shall be classified HSS. In addition to being HSS in terms of their contribution to CDF or LERF, piping segments might also be HSS in terms of other risk metrics or conditions. Therefore, the following conditions shall be evaluated. Piping segments determined to be Medium consequence category in any table by the consequence evaluation (I-3.1.1) and (I-3.1.2) shall be determined HSS or LSS by considering the RISC evaluation and the other relevant information (I-3.1.3, I-3.1.4, and I-3.1.5) provided for determining classification.	Objection requiring qualification	Piping segments determined to be Medium, Low or None (no change to base case) consequence category in any table by the consequence evaluation in Section I-3.1 shall be determined HSS or LSS by considering the other relevant information for determining classification. The following conditions shall be evaluated and answered true or not true. **If any of the following ~~above~~ eleven (11) conditions are not true, HSS should be assigned.** . . ~~If any of the above eleven (11) conditions are not true, HSS should be assigned.~~
{I-3.1.3(a)(1)} [ ]	Failure of the piping segment will significantly increase the frequency of an initiating event, including those initiating events originally screened out in the PRA, such that the CDF or large early release frequency (LERF) would be estimated to increase by more than 10-6/yr or 10-7/yr, respectively.	No objection	*Consideration may be deleted because additional quantitative risk guidelines are unnecessary when passive classification is performed based on consequences*
{I-3.1.3(a)(2)} [ ]	Failure of the piping segment will compromise the integrity of the reactor coolant pressure boundary as defined in –1200(b).	No objection	*Consideration may be deleted because -1200(b) is retained and already assigns the same piping to the HSS category.*

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.3(a)(3)} [I-3.2.2(b)(1)]	Even when considering operator actions used to mitigate an accident, failure of the piping segment will fail a high safety significant function.	Objection requiring qualification	Even when taking credit for plant features and operator actions, failure of the piping segment will not ~~directly~~ fail ~~another~~ high safety-significant function.
{I-3.1.3(a)(4)} [I-3.2.2(b)(2)]	Failure of the piping segment will result in failure of other safety-significant piping segments, e.g., through indirect effects.	Objection requiring qualification	Failure of the piping segment will not result in failure of ~~another~~ high safety-significant piping segment, e.g., through indirect effects.
{I-3.1.3(a)(5)} [I-3.2.2(b)(3)]	Failure of the piping segment will prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions.	Objection requiring qualification	Consideration changed and moved to new Section I-3.2.2(b)(3), ~~Even when taking credit for plant features and operator actions~~, failure of the piping segment will not prevent ~~or adversely affect~~ the plant~~'s capability to reach or maintain~~ reaching or maintaining safe shutdown conditions.
{I-3.1.3(b)(1)} [ ]	The piping segment is a part of a system that acts as a barrier to fission product release during severe accidents.	No objection	*Consideration may be deleted. The original guidance is excessively conservative. Once the excessive conservatism is removed, the response to this consideration is not expected to identify any piping as HSS piping that would not be assigned HSS by the CLERP related guidelines.*
{I-3.1.3(b)(2)} [I-3.2.2(b)(4)]	The piping segment supports a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines.	No objection	The piping segment does not individually support a sole means for successful performance of operator actions addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines required to mitigate an accident or transient, including instrumentation and other equipment associated with the required actions.

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.3(b)(3)} [I-3.2.2(b)(6)]	Failure of the piping segment will result in unintentional releases of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100.	Objection requiring qualification	~~Even when taking credit for plant features and operator actions, f~~Failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions.

{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.4} [I-3.2.2(b) (7-11)]	Maintain Defense in Depth. When categorizing piping segments LSS, the RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant Specific Changes to the Licensing Basis." Dated July 1998.	No objection	The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated November 2002. Defense-in-depth is maintained if: (7) A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. (8) Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided. (9) 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). (10) Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed. (11) Independence of fission-product barriers is not degraded.

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.1.5} [I-3.2.2(c)]	Maintenance of Adequate Safety Margins. When categorizing piping segments LSS, the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety 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 safety 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.	No objection	If LSS has been assigned from I-3.2.2(b), then the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety 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 safety margins are maintained by: (1) Ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or (2) Ensuring that proposed revisions account for analysis and data uncertainty. If LSS has been assigned from I-3.2.2(b) and at least one of the above safety margin conditions are true, then LSS should be assigned; if both of the above safety margin conditions are not true, then HSS shall be assigned.
{I-3.2} [I-3.2]	I-3.2 Classification	Objection requiring qualification	I-3.2 Classification Risk Informed Safety Classification is determined by considering the Consequence Category ~~in conjunction with other relevant information~~.

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{N-660, R0 Section} [WCAP-16308 Section]	ASME Code Case N-660 Revision 0	Position	Resolution of WCAP-16308 Section WCAP-16308 Changes to Code Case N-660 Text not marked NRC Staff additions in Bold, deletions in Strikeout, and comments in Italic
{I-3.2.2(b)} [I-3.2.2(b)(5)]	Any piping segment initially determined to be a Medium consequence category and that is subject to a known active degradation mechanism shall be classified HSS.	No objection	The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program).
{ } [I-3.2.2(b) footnote]		No objection	To credit operator actions, the following criteria must be met: • There must be an alarm or clear indication of the failure. • A procedure must direct the response to the alarm or indication. • Equipment activated to alleviate the condition must not be affected by the failure. There must be sufficient time to perform the compensatory action.
{Table I-1 row "I"} [Table I-1 row "I"]	N/A	No objection	None

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Note 1 – Figure I-1, Risk-Informed Safety Classification Process

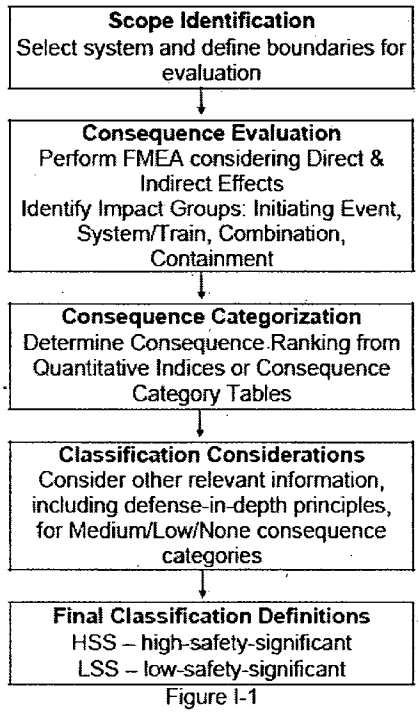

Figure I-1
Risk-Informed Safety Classification Process

RESOLUTION OF COMMENTS ON DRAFT SAFETY EVALUATION (SE) FOR

TOPICAL REPORT (TR) WCAP-16308-NP, REVISION 0

"PRESSURIZED WATER REACTOR OWNERS GROUP 10 CFR [TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS*] 50.69 PILOT PROGRAM –

CATEGORIZATION PROCESS - WOLF CREEK GENERATING STATION [(WCGS)]"

NUCLEAR ENERGY INSTITUTE (NEI)

PROJECT NO. 689

By letter dated October 14, 2008, the NEI provided three substantive comments on the draft SE for TR WCAP-16308-NP, Revision 0. The following are the U.S. Nuclear Regulatory Commission (NRC) staff's resolution of these comments.

NEI Comment #1: Credit for Operator Actions

On page 5 of the Draft SE at lines 28 through 34, the NRC staff's technical evaluation of operator actions can be misinterpreted by licensees with respect to the types of operator procedures and guidance that can be credited for operator actions. Emergency procedures, some of which are very prescriptive if-then instructions requiring verbatim compliance by the operators, are frequently referenced by symptom-based procedures. When these human actions are quantified in a probabilistic risk assessment using acceptable human error analyses methods, they are usually shown to be highly reliable. We believe that the NRC staff position is to limit the types of procedures and guidance to only those that result in well defined and predictable actions rather than any operator actions. Therefore, it is proposed that the paragraph on page 5 at lines 28 through 35 be changed to provide this clarification as follows:

> The TR argues that its proposal only permits credit if a procedure directs the operators' response. While some plant procedures provide very prescriptive if-then instruction to the operators, other plant procedures and guidance may direct the operators, in general, to develop and attempt mitigative actions. In the latter case, any conceivable mitigative actions would satisfy the criterion. The NRC staff only accepts the proposed credit for operator actions when it can be shown that the actions that have a high likelihood of success, e.g., well defined and predictable actions. Qualitatively crediting actions with a low likelihood of success could place high-safety significant (HSS) structures, systems, and components (SSCs) into low-safety significant (LSS).

In addition, we would propose a similar clarification to the "Limitations and Conditions" on page 10 at lines 33 through 37 as follows:

> 3. As described in Section 3.2 of the SE, the NRC staff only accepts credit for operator actions that have a high likelihood of success, e.g., well defined and predictable actions. Qualitatively crediting actions with a low likelihood of success could place HSS SSCs into LSS.

ATTACHMENT 2

- 2 -

Finally, it is proposed that the entries regarding operator actions in Table 1 of the SE at pages 8 and 9 should be changed to state:

For the row beginning with {I-3.1.3(a)(5)} [I-3.2.2(b)(3)]:

> Consideration changed and moved to new Section I-3.2.2(b)(3), "Even when taking credit for plant features and highly reliable operator actions, failure of the piping segment will not prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions."

For the row beginning with {I-3.1.3(b)(3)} [I-3.2.2(b)(6)]:

> Even when taking credit for plant features and highly reliable operator actions, failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions.

NRC Response:

The NRC staff does not agree with the NEI proposed changes regarding operator actions. The operator actions in the questions would need to be very highly reliable to yield categorization that is consistent with the quantitative guidelines. The staff does not agree that the generic guidance provided in the TR provides confidence that only very highly reliable actions will be credited. The comments provided by NEI do not address a factual error but the staff has expanded is discussion in Section 3.2 of the SE to clarify its reasons for not accepting the original changes proposed in the TR.

NEI Comment #2: Monitoring of Risk-Informed Safety Class (RISC)-3 SSCs

Page 8 of the draft SE, at lines 35 through 45, states the following:

> The NRC staff finds that the information provided in TR WCAP-16308-NP and in the RAI response does not provide a sufficient basis for assuming that the regulatory requirements of 10 CFR 50.69(d)(2)(i) or (d)(2)(ii) would be satisfied. The monitoring of RISC-3 SSCs appears to be primarily focused on the monitoring of SSC failures and does not allow the NRC staff to conclude that degradation of RISC-3 SSCs would be monitored and corrected in a manner that will provide reasonable confidence that these SSCs would remain capable of performing their safety-related functions under design-basis conditions. For example, specific information on periodic inspections and tests that could be used to detect and correct degradation of RISC-3 SSCs was not provided. Therefore, the NRC staff cannot reach a finding that the monitoring of RISC-3 SSCs as described in Section 7.3 of TR WCAP-16308-NP will result in the required degree of "reasonable confidence" to satisfy 10 CFR 50.69(d)(2).

We believe the above paragraph should be clarified to provide factual consistency with NRC Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance." This RG provides formal NRC positions on the categorization methodology provided by NEI 00-04. Regulatory position 7 states the following (emphasis added):

- 3 -

Common-Cause Failure and Degradation Mechanism Considerations in Revision 0 of NEI 00-04

> The NRC staff notes that mechanisms that could lead to large increases in core damage frequency (CDF) and large early release frequency (LERF), which could potentially invalidate the assumptions underlying the categorization process, including the risk sensitivity study, are the emergence of extensive common-cause failures (CCFs) impacting multiple systems and significant unmitigated degradation. However, for these types of impacts to occur, the mechanisms that lead to failure, in the absence or relaxation of treatment, would have to be sufficiently rapidly developing or not self-revealing, such that there would be few opportunities for early detection and corrective action. Section 12.4 of NEI 00-04 describes an acceptable performance-based approach to address these concerns.
>
> Alternatively, those aspects of treatment that are necessary to prevent significant SSC degradation or failure from known mechanisms, to the extent that the results of the risk sensitivity study would be invalidated, could be identified by the licensee or applicant, and such aspects of treatment would be retained. This alternative approach would require an understanding of the degradation and common-cause failure mechanisms and the elements of treatment that are sufficient to prevent them.

The paragraph quoted above from the draft SE for the TR is based on the assumption that Section 12.4 of NEI 00-04 is, in itself, insufficient to address degradation mechanisms, and that the alternative method described in the RG is in fact mandatory. This contradicts the position of the RG, which clearly states the acceptability of the performance-based NEI method and the alternative nature of the programmatic method. We request that the Final SE be clarified to be consistent with regulatory position 7 of RG 1.201. One method to accomplish this would be to delete the second and third paragraphs of Section 3.5 of the SE, and to affirm the acceptability of the NEI 00-04 method.

Corresponding Section 5.2 of the draft SE, at lines 33 through 43 should also be clarified, or removed, for the reasons stated above.

NRC Response:

RG 1.201 does not discuss or approve any treatment method, but only discusses monitoring as it relates to updating the categorization process. RG 1.201 discusses CCFs and degradation, and that measures need to be taken to address them when implementing the guidance contained in Section 12.4 of NEI 00-04 related to the categorization process. Section 7.3 of TR WCAP-16308-NP, refers to periodic updates of the categorization process based on results of monitoring, and does not discuss specific treatment methods used to provide the monitoring necessary to meet this function.

The NEI comments on the draft SE imply that there is a conflict between the SE and RG 1.201. This most likely was caused by specific references to aspects of treatment in Section 3.5 of the SE, such as references to 50.69(d)(2)(i) and (ii) the "alternative treatment" section of the rule. Although the NRC staff observations in Section 3.5 are factually correct, part of the discussion was more pertinent to the treatment method than to the update of categorization based on monitoring. To avoid confusion, the NRC staff revised Section 3.5 and its corresponding Section 5.2 of the SE to place more emphasis on the specific role of monitoring with respect to

categorization adjustments. The NRC staff, however, did retain elements related to monitoring and corrective action to ensure that degradation of RISC-3 SSCs is corrected in a manner that supports "reasonable confidence" of operability as required by the rule. This revision does not constitute an endorsement of the monitoring aspects of the treatment method since no plant-specific treatment plan was provided.

NEI Comment #3: Application of RISC-3 Treatment Requirements

The NRC Draft SE, page 9, at lines 10 through 20, states the following:

> The NRC staff finds that the general information provided in TR WCAP-16308-NP and in the RAI response does not provide a sufficient basis for determining that the regulatory requirements of 10 CFR 50.69(d)(2) would be satisfied. The lack of a more specific description of the treatment of RISC-3 SSCs at WCGS prevents the NRC staff from reaching a determination that reasonable confidence exists that RISC-3 SSCs will remain capable of performing their safety-related design basis functions, and that the treatment will be consistent with the categorization process. One example of an acceptable description of treatment to be applied to safety-related low safety significant SSCs is provided in the NRC SE dated August 3, 2001, that accepted the request by the South Texas Project (STP), Units 1 and 2, for exemption from special treatment requirements specified in certain NRC regulations.

Paragraph (b)(2) of 10 CFR 50.69 provides a discussion of the information that must be submitted by a licensee to implement the rule. As correctly noted in the draft SE, the licensee is not required to submit their plan for treatment of SSCs to the NRC for review and approval under this provision of the rule. There are no other requirements for submittal content beyond those specified in paragraph (b)(2). In accordance with this approach, the NRC has not developed regulatory guidance addressing treatment of RISC-3 SSCs. The industry, through EPRI, has developed such guidance for industry's use in implementing the rule.

Paragraph (d)(2) of 10 CFR 50.69 addresses the need for RISC-3 SSCs to be subject to inspection, testing and corrective action to provide reasonable confidence of performance under design basis conditions. Licensees implementing the rule must conform to these requirements; however, the rule does not require description of these programs as part of the license amendment request. Therefore, while the TR does not contain specific descriptions of these treatment methods, there is no expectation that it should. The SE references information provided in the Final Safety Analysis Report for STP in their implementation of an exemption to the special treatment rules. However, 10 CFR 50.69 is structured differently than the STP exemption, in that there were no high level regulatory requirements (e.g., 10 CFR 50.69 (d)(2)) for treatment of low risk SSCs available for STP. Therefore, this reference is not pertinent. The NRC staff does not need to make a finding of conformance to these rule provisions as part of a license amendment request, nor as part of their approval of the TR.

Given that the SE concludes that no description of treatment is necessary, we believe the extensive discussion of RISC-3 treatment in Section 3.6 of the SE is not necessary and will create potential confusion due to its contradiction with the rule. Section 5.3 of the SE, which contains a similar discussion of RISC-3 SSC treatment, should be similarly changed or omitted.

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NRC Response:

Section 3.6 was modified to emphasize the fact that licensees are not required to submit their plan for SSC treatment to the NRC for review and approval, but that nonetheless, RISC-3 SSC treatment, in the form of inspection, testing and corrective action, must provide reasonable confidence that these SSCs will continue to perform their safety-related functions under design basis conditions as required by 10 CFR 50.69(d)(2).

The fact that licensees are not required to submit treatment plans as part of a 10 CFR 50.69 application does not eliminate the need for the NRC staff to make a finding on the treatment section that was included in TR WCAP-16308-NP. Specifically, the staff cannot give licensees implementing 10 CFR 50.69 the impression that a simple reference to application of commercial practices and monitoring of failure rates would constitute an adequate plan for treatment of RIC3 SSCs. Based on the minimal amount of information provided in the TR, the NRC staff cannot make a conclusion that the proposed treatment plan would be acceptable.

In addition, the discussion of the South Texas Project (STP) exemption from special treatment requirements specified in certain NRC regulations was removed from Section 3.6 of the SE. It was originally provided as one example of an acceptable description of treatment to be applied to safety-related low safety significant (LSS) SSCs that was evaluated by the NRC in an SE dated August 3, 2001. In this evaluation, which pre-dates the 10 CFR 50.69 rule, the NRC staff reviewed the elements and high-level objectives of the treatment processes for safety-related LSS SSCs specified by STP in a proposed revision to its Final Safety Analysis Report (FSAR). The proposed FSAR revision provided a high-level description of eight treatment processes (design control; procurement; installation; maintenance; inspection, test, and surveillance; corrective action; management and oversight; and configuration control) intended to provide reasonable confidence that safety-related LSS SSCs will maintain their functionality under design-basis conditions. In the SE, the NRC staff concluded that the treatment processes described in the proposed FSAR revision for STP contained elements and high-level objectives that, if effectively implemented, will provide reasonable confidence that safety-related LSS SSCs are capable of performing their safety functions under design-basis conditions, including environmental and seismic conditions, throughout their service life. Although the scope of treatment elements described in the STP exemption exceeds the scope specified in 10 CFR 50.69(d)(2) (inspection, testing and corrective action), this example was cited as one acceptable approach to RISC-3 treatment, albeit one that exceeds the minimum requirements set forth in 10 CFR 50.69(d)(2). In the absence of other regulatory guidance, it may be useful as a framework for describing certain elements of a RISC-3 treatment program that would satisfy 10 CFR 50.69. The staff acknowledges that the STP approach goes beyond that which is required by 10 CFR 50.69(d)(2), and is not requiring that licensees electing to implement 10 CFR 50.69 adopt the approach to RISC-3 treatment described in the STP exemption.

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Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)		X
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)		X
Dominion Connecticut	Millstone 3 (W)		X
Dominion Kewaunee	Kewaunee (W)		X
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)		X
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	X	
Duke Energy	Oconee 1, 2, 3 (B&W)		X
Entergy	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)		X
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W), TMI 1 (B&W)	X	
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W)	X	
	Davis-Besse (B&W)		X
Florida Power & Light Group	St. Lucie 1 & 2 (CE)		X
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)		X
Florida Power & Light Group	Pt. Beach 1&2 (W)		X
Luminant Power	Comanche Peak 1 & 2 (W)	X	
Xcel Energy	Prairie Island 1&2 (W)	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W),	X	
	Crystal River 3 (B&W)		X
PSEG – Nuclear	Salem 1 & 2 (W)	X	
Southern California Edison	SONGS 2 & 3 (CE)		X
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

- **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PA-SEE-0027

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B		X
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3		X
Hokkaido	Tomari 1 & 2 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 1, 2 & 3, Ohi 1, 2, 3 & 4, Takahama 1, 2, 3 &4 (W & MHI)		X
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)		X
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)		X
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X
Electricite de France	54 Units		X

- **This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

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Acronyms	
AOV	Air Operated Valve
ASME	ASME (formerly known as the American Society of Mechanical Engineers)
CCF	Common Cause Failure
CFR	Code of Federal Regulations
EN	Wolf Creek designation for the Containment Spray System
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
FIVE	Fire Induced Vulnerability Evaluation
GK	Wolf Creek designation for the Control Building Ventilation System
HEP	Human Error Probability
HSS	High Safety Significant
IDP	Integrated Decision-making Panel
IPEEE	Individual Plant Examination for External Events
LSS	Low Safety Significant
MOV	Motor Operated Valve
MR	Maintenance Rule
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PAG	Protective Action Guides
PRA	Probabilistic Risk Assessment
PWROG	Pressurized Water Reactor Owners Group (formerly Westinghouse Owners Group)
RI-ISI	Risk-Informed In-Service Inspection
RISC	Risk Informed Safety Classification
SAMG	Severe Accident Management Guideline
SSC	Systems, Structures and Components
SSEL	Safe Shutdown Equipment List
SMA	Seismic Margins Analysis
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation
WinNUPRA	A computer code used to model the PRA

1. INTRODUCTION AND BACKGROUND

1.1 INTRODUCTION

The new 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," is a voluntary regulation that provides an approach by which licensees can categorize safety-related and non-safety-related structures, systems and components (SSCs) according to their safety significance. The intent of 50.69 is to provide a means for appropriately focusing attention on those SSCs that are most important to safety, while maintaining a high degree of confidence that all SSCs will be capable of performing their design basis functions. To achieve this, 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. Per 50.69 (b)(2), a licensee desiring to take advantage of this voluntary regulation would make a one-time submittal to the Nuclear Regulatory Commission (NRC) for review and approval. In accordance with 50.69 (b)(2), the submittal is to contain documentation related to: a) the categorization process to be used to define the safety significance of SSCs for which treatment could be relaxed, b) the process used to establish the technical capability of the licensee's Probabilistic Risk Assessment (PRA) that is used as a basis for the categorization, c) the process used to assure that adequate safety margins are maintained and the potential increases in risk resulting from changes in treatment permitted by 50.69 are small. Upon NRC review and approval of these processes, the licensee could then apply the categorization process to as many (or as few) systems as desired, provided that an entire system is considered. Although the proposed 50.69 rule also contains requirements for assuring that any changes in treatment for low safety significant SSCs do not result in unacceptable changes in margins or risk, there is no requirement for the licensee's treatment process to be included in the submittal for NRC review and approval. The effectiveness of the treatment processes would be assured through NRC inspections and performance requirements as described in the 50.69 rule.

The objective of this report is to provide documentation of the categorization process used by Wolf Creek Nuclear Operating Corporation (WCNOC) in support of a future licensee submittal requesting approval to implement 50.69 at the Wolf Creek Generating Station (WCGS). A separate report on the quality of the PRA to be used in the categorization process will be submitted to the NRC by WCNOC to support the license amendment request to implement 50.69 at WCGS. Since the categorization process, as opposed to the categorization results, is the subject of the NRC review and approval, the two can be reviewed and approved separately. That is, approval for implementation of 50.69 requires a robust categorization process AND a technically capable PRA to provide input to the categorization process. However, both are independent of each other and can be reviewed separately without compromising the adequacy of the review. Thus, review and approval of the categorization process, as described in this report, can be completed without a coincident review of the PRA.

While the categorization process described in this report has been applied to two systems at WCGS, it is recognized by WCNOC that the results of the categorization cannot be directly used in the implementation of 50.69 since the quantitative risk assessment was not based on a PRA that meets the technical adequacy requirements of the 50.69 rule. Following future NRC approval of the PRA technical adequacy, per 50.69 (b)(2), the categorization results for these two systems will be re-visited to finalize the SSC categorization.

The categorization process to be used by WCNOC was developed based on categorization of two systems at WCGS. The Containment Spray System (EN System) represents a

stand-by emergency system whose primary function is to mitigate the consequences of an accident by reducing containment pressure and scrubbing fission products released from the core that are airborne in the containment. The Control Building Ventilation System (GK System) is a normally operating system with some additional post-accident functions whose primary function is cooling of various components in the control building and providing a habitable environment in and near the control room. These systems were chosen, in part, based on their absence from the PRA as a result of various PRA assessment and screening processes. Thus, the categorization of these systems identifies the processes to be used for categorization when there is little or no PRA information to use as a quantitative basis or where risk impacts must be inferred from components that are modeled in the PRA.

The categorization of the structures, systems, and components documented in this report were performed in accordance with the final draft version of NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), as transmitted to the NRC by the Nuclear Energy Institute in April of 2004, and a proposed Revision 2 to the ASME N-660 Code Case, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities." Differences between the categorization guidelines in those reports and the most recent versions of those guidelines that are endorsed by the NRC in Regulatory Guide 1.201 (Reference 2) and Revision 14 to Regulatory Guide 1.147 (Reference 3) are discussed in Appendix A of this report.

1.2 BACKGROUND

The intent of the 10 CFR 50.69 regulatory initiative is to adjust the scope of equipment subject to special regulatory treatment (controls) to better focus attention and resources on equipment that is safety significant. The implementation of 10 CFR 50.69 does not replace the existing "safety-related" and "nonsafety-related" categorizations. Rather, it divides these categorizations into two subcategories based on high or low safety significance as depicted in Figure 1-1. This is called the Risk-Informed Safety Classification (RISC) process.

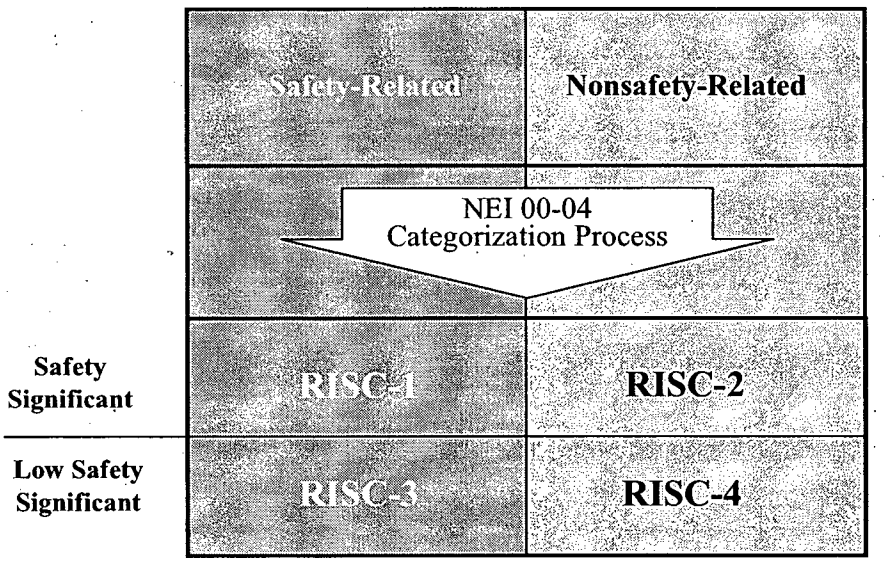

Figure 1-1 Risk-Informed Safety Classifications (RISC)

During the 50.69 rulemaking process, the industry developed two sets of categorization guidance to assist licensees in implementing 50.69. The first set of guidance, developed under the auspices of the Nuclear Energy Institute (NEI), is for use in categorizing the active functions of SSCs. The second set of guidance, developed under the auspices of ASME, is for use in categorizing the passive, or pressure boundary, functions of SSCs. Both sets of guidance have been extensively tested by pilot applications of the 50.69 process prior to the issuance of the final rule in November of 2004. The guidance has been improved as a result of each pilot activity.

Both sets of guidance use an integrated decision-making process to define the scope of equipment that will be subject to the special treatment provisions in the regulations. This integrated decision-making process blends risk insights, deterministic considerations, and operational feedback through the involvement of a group of experienced plant personnel representing diverse plant functions and responsibilities.

The NEI 00-04 guidance requires that the final decision regarding the categorization of components performing an active function functions be presented to an expert panel, known as the Integrated Decision-making Panel (IDP). The IDP is composed of a group of experienced plant personnel representing diverse plant functions and responsibilities. The IDP is supported by additional working level groups of personnel to provide detailed investigations and assessments necessary for IDP consideration of the categorization of the SSCs under consideration. A preliminary categorization is performed by a working group consisting primarily

of working level personnel (e.g., engineers, plant operators, etc.). This preliminary categorization is then presented to the IDP, which is composed of qualified professionals and supervisory personnel who are selected by plant management to consider the information and come to a final decision. The NEI 00-04 guidance also provides guidance for the purpose, composition, and conduct of the IDP.

ASME guidance is contained in Code Case N-660 (Reference 4). Code Case N-660 also prescribes the use of an expert panel to finalize the scope of pressure boundary components (and their supports) that will be subject to ASME special treatment provisions. As with the assessment of active components described in NEI 00-04, this process blends risk insights, deterministic considerations, and operational feedback. Since the ASME Code Case N-660 also requires the involvement of a group of experienced plant personnel representing diverse plant functions and responsibilities, both the preliminary engineering categorization according to NEI 00-04 and ASME Code Case N-660 were presented at one IDP session. Performing the final categorization for both active and pressure boundary components for a given system at the same IDP session helps to ensure consistency in the IDP considerations for both active and pressure boundary components. However, due to differences in the process of grouping components between the NEI and ASME guidance, the IDP considerations for active and pressure boundary components were not done at the same time. Each IDP session on a given system first considered all of the components active functions in the first part of the IDP session. The pressure boundary functions were then considered in the second part of the IDP session.

2. WOLF CREEK GENERATING STATION PILOT PROGRAM SCOPE & APPROACH

2.1 INTRODUCTION

The preliminary categorization of SSCs for the selected systems was completed in the latter half of 2003, using the draft NEI 00-04 guidance available at that time and the approved ASME Code Case N-660, Revision 0. The initial WCGS Integrated Decision-making Panel (IDP) to consider the initial categorization recommendations was held in December of 2003. That IDP identified a number of issues with the use of the NEI and ASME guidance documents. Most of the issues involved the possibility of multiple interpretations for certain guidance elements. In order to achieve stability in the categorization process, the December 2003 IDP tabled all matters and requested that the affected guidance elements be revised to resolve the issues identified by the IDP. Following revisions to the NEI and ASME guidance, the IDP was reconvened in April of 2004. This IDP session was a complete review of all SSCs for both systems and did not rely upon any recommendations made at the initial IDP session. Following the completion of the second IDP, the minutes were documented and archived. The development of this submittal report was delayed pending finalization of the 50.69 rule and the resolution of all major industry and regulatory comments on both the NEI and ASME guidance documents. An earlier submittal of this report for regulatory review would have made the review difficult because an endorsed set of guidance would not have been available to use as a standard for the regulatory review.

The following sections discuss the process to be used at WCGS to categorize SSCs for 50.69 implementation. The process followed the NEI and ASME guidance for categorization of SSCs that was available in April of 2004. Appendix A provides a discussion of the differences between the 2004 guidance and the guidance that has been endorsed by NRC in Regulatory Guides 1.201 and 1.147, Revision 14 and the impact of changes in the WCGS categorization process and/or results.

2.2 SCOPE OF SSCS SELECTED FOR §50.69 CATEGORIZATION

Two systems were selected to define and validate the categorization process to be used at WCGS for the implementation of the proposed 10 CFR 50.69:

- Containment Spray System
- Control Building Ventilation System

The Containment Spray System (EN System) represents a stand-by emergency system whose primary function is to mitigate the consequences of an accident by reducing containment pressure and scrubbing airborne fission products released from the core. The EN system is composed of the components required to deliver water to the containment via the containment spray headers from either the Refueling Water Storage Tank (RWST) or the containment sump, and those components required to provide sodium hydroxide to buffer the pH of the containment sump water.

The Control Building Ventilation System (GK System) is a normally operating system whose primary function is cooling of various components in the control building and providing a habitable environment (i.e., temperature) in and near the control room. In the post accident mode, some of the ventilation flow paths are altered to provide primary functions associated with control room habitability (both temperature and radiation) and to provide cooling for class 1E components (i.e., switchgear) in the control building that support accident mitigation functions.

These systems were chosen to demonstrate the categorization process to be used at WCGS for implementation of 50.69, based in part, on their limited modeling in the PRA as a result of various PRA assessment and screening processes. The categorization of system functions whose components are explicitly modeled in PRA is straight-forward using the NEI and ASME guidance. The categorization of these systems identifies the processes to be used for categorization when there is little or no PRA information to use as a quantitative basis for categorizing system functions and components.

2.3 APPROACH

The NEI guidance described in NEI 00-04 was used in the categorization of the active functions of the SSCs in the EN and GK systems for WCGS. The April 2004 version of NEI 00-04 was used for the pilot evaluation described in this report. The manner in which WCNOC applied NEI 00-04 to the categorization of SSCs at WCGS is described in Section 3 of this report.

The classification of SSCs having a pressure retaining function (also referred to as passive components) was performed using an April 2004 draft revision of ASME Code Case N-660. The Code Case N-660 process may be applied to any of Class 1, 2, 3, or non-class pressure-retaining items or their associated supports, except core supports. Specifically excluded from the risk informed safety classification process in Code Case N-660 are reactor coolant system pressure boundary components (which make up the majority of the Class 1 pressure boundary components in a PWR). The regulatory position in Revision 14 to Regulatory Guide 1.147, that endorses Code Case N-660, Revision 0, takes exception to its use for <u>any</u> Class 1 pressure retaining items. No Class 1 pressure retaining items were found in the EN or GK Systems, so this exception was not addressed in the present categorization. For future use of the categorization process described in this report, WCNOC will not consider re-classification of any Class 1 pressure retaining item unless endorsed by NRC in further revisions to Regulatory Guide 1.147. The manner in which WCNOC applied Code Case N-660 to the categorization of SSCs at WCGS is described in Section 4 of this report.

3. CATEGORIZATION OF ACTIVE FUNCTIONS

The overall process used in categorizing the active functions and components, as described in NEI 00-04, is depicted in Figure 3-1. This process builds upon the insights and methods from many previous categorization efforts, such as the risk ranking in the risk-informed in-service inspection (RI-ISI), Maintenance Rule (MR), motor operated valve (MOV) and air operated valve (AOV) programs. It is a comprehensive, robust process that includes consideration of various contributors to plant risk and defense-in-depth.

The process includes eight main steps:

- Assembly of Plant-Specific Inputs
- System Engineering Assessment
- Component Safety Significance Assessment
- Defense-In-Depth Assessment
- Preliminary Engineering Categorization of Functions
- Risk Sensitivity Study
- IDP Review and Approval
- SSC Categorization

The manner in which each of these steps was implemented for the categorization of SSCs in the EN and GK systems for WCGS are discussed in the following sections. The process used for the GK and EN systems, as described in the following sections, is the basis for implementation of 50.69 for other systems at WCGS.

3.1 ASSEMBLY OF PLANT-SPECIFIC INPUTS

Section 3 of the NEI 00-04 guidance discusses process for the collection and assessment of the necessary inputs to the risk-informed categorization process to fulfill the requirements of paragraphs 50.69(b)(2)(ii) and (iii) and 50.69(c)(1)(i). This includes design and licensing information, PRA analyses, and other relevant plant data sources, as well as the critical evaluation of plant-specific risk models to assure that they are adequate to support this application.

WCNOC Implementation

The WCNOC assembly of plant specific inputs followed the guidance, without exception, in Section 3 of NEI 00-04. The details of the implementation are described in the following paragraphs.

The sources of information related to the plant design and operation that are used as inputs in the WCGS SSC categorization process are listed in Table 3-1. The intent of Table 3-1 is to show that all of the inputs to the 50.69 categorization are controlled processes at WCGS and have administrative procedures to define the control processes.

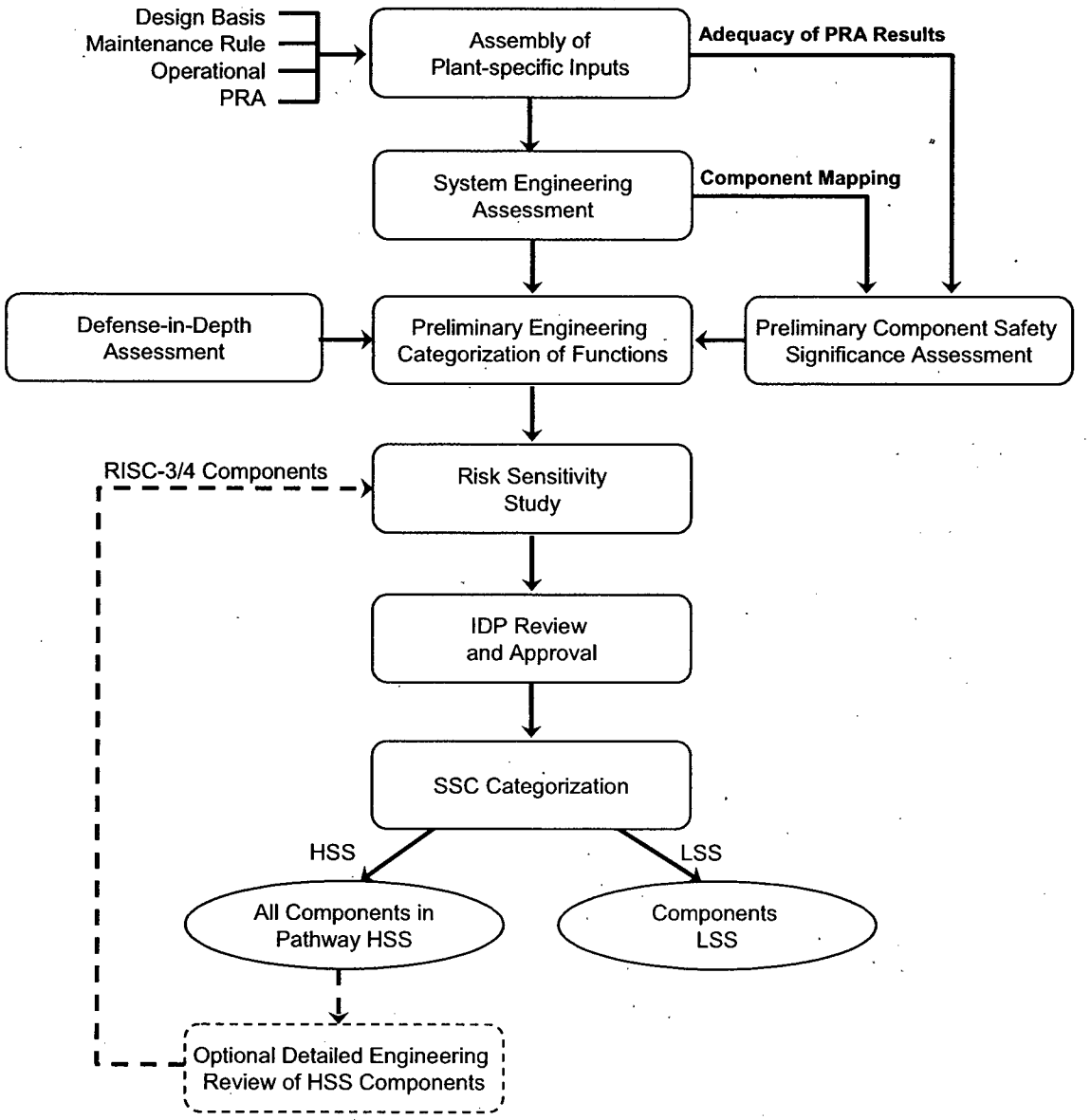

Figure 3-1: Risk-Informed Categorization Process (Adapted from NEI 00-04)

Table 3-1: Design and Operational Inputs to the WCGS Categorization Process		
Inputs	**Controlled Process**	**Admin Procedure**
Asset Management List	✓	✓
Tag Numbers	✓	✓
Design Basis Functions	✓	✓
Process and Instrumentation Diagrams (P&IDs)	✓	✓
Maintenance Rule Functions	✓	✓
System Descriptions	✓	✓
Risk Informed ISI	✓	✓
Probabilistic Risk Assessment (PRA)	✓	✓
Individual Plant Examination for External Events (IPEEE)	✓	✓
Safe Shutdown Pathway Determination	✓	✓

The identification of the components in each system is taken from the WCGS Asset Management List which identifies all of the safety-related and non-safety-related SSCs in the plant, including identification of augmented quality SSCs. The Asset Management List includes a component identifier (i.e., Tag Number), the associated system, component type, component description, and safety class. All components in the WCGS are identified by a unique Tag Number that identifies the system to which the component belongs, the type of component (e.g., check valve) and its unique identifying number.

For each system categorized, all components in the Asset Management List with a system identifier (asset tag) matching the system being categorized are included in the scope of each system categorization. Thus, the Asset Management List serves as the definition of the system boundaries for 50.69 assessments.

As part of the process, all system components belonging to the selected system are identified on the P&IDs associated with the associated system to further validate the Asset Management List. This mapping of components to the P&IDs also serves to familiarize the engineering staff doing the preliminary categorization with the location of the component in the system to facilitate the functional mapping described below.

The WCGS Maintenance Rule (MR) functions and Design Basis (DB) functions are used to define a set of active functions for use in the 50.69 categorization of a system. These functions are later used in the component mapping process.

The system descriptions for the associated systems are then used to assist in the identification of the "flow paths" on the P&IDs that represent each of the system functions.

The WCGS model used for this analysis was the 1998 internal events model, called Revision 4 (Reference 5). The results have been provided as input to this assessment. The Individual Plant Examination for External Events (IPEEE), has been used for the fire, seismic, other

external events risk assessments (Reference 6). The shutdown risks are taken from the safe shutdown equipment list.

The PRA input used for the RISC of the components in the EN and GK systems has not been updated to reflect the findings of the PRA peer review nor other quality assessments as described in Regulatory Guide 1.200. Thus, a separate submittal related to the PRA technical adequacy will be made by WCNOC, when the decision is made to implement 50.69, to satisfy the PRA requirements of 50.69. However, the input from the 1998 PRA model is adequate to define the categorization process to be used at WCGS. Following the completion of any PRA modifications that are required to meet the PRA technical adequacy requirements for use in the proposed 10 CFR 50.69 categorization, the present RISC results, which reflect the risk insights from the current PRA, will be updated as appropriate to assure that the results of the categorization process reflect the updated PRA.

3.2 SYSTEM ENGINEERING ASSESSMENT

Section 4 of the NEI 00-04 guidance discusses the requirements for the initial engineering evaluation of a selected system to support the categorization process to fulfill the requirements of paragraph 50.69(c)(1)(i). This includes the definition of the system boundary to be used and the components to be evaluated, the identification of system functions, and an initial mapping of components to functions. The system functions are identified from a variety of sources including design/licensing basis analyses, Maintenance Rule assessments and PRA analyses. The mapping of components is performed to allow the correlation of PRA importance measures to system functions.

WCGS Implementation

The WCNOC system engineering assessment followed the guidance, without exception, in Section 4 of NEI 00-04. The details of the implementation are described in the following paragraphs.

Definition of System Boundaries

Preliminary system boundaries were determined from the system component list for each system provided from the WCGS Asset Management List. Any component on the Asset Management List with a system identifier matching the system being categorized (i.e., EN or GK) is included within the scope of the system categorization. The major components on the system equipment list are located on the system P&IDs to further verify the list.

Any discrepancies between the system equipment list and the system diagrams are resolved before proceeding further. For example, a limited number of components may be included on the Asset Management List that are not on the system P&IDs. In many cases, these components are piece parts of components already identified on the P&IDs. Congruence between the two sources of information is sought in all cases to validate the Asset Management List. The Asset Management List for the system is further verified, and modified if necessary, as part of the mapping process described below. This list serves as the definition of the system boundaries for this system categorization.

The system boundaries where components within the system being categorized interface with another system are especially important. In some cases, these interfacing components cannot be completely categorized because their impact on the interfacing system is unknown at the time of categorization of the system to which they are assigned based on their tag numbers. Some examples include:

- Electrical components may not be completely categorized where the isolation device is not part of the system being categorized. In this case, the impact of a failure of the component

in the system being categorized on the remainder of the electrical system cannot be determined. This can apply to power circuits, control circuits and instrumentation circuits. In cases where the isolation device can be identified, even though it is not part of the system being categorized, the assessment can be completed, but the basis for the importance of the isolation device must be documented and carried forward to the categorization of the interfacing system.

- In certain instances, the component tag may indicate that the component belongs to the system being categorized, but it is really part of a support or interfacing system. An example of this is the support system piping connected to heat exchangers. When the heat exchanger is part of the system being categorized, some of the piping and instrumentation associated with the interfacing system side of the heat exchanger may also be assigned to the system being categorized even though it more appropriately belongs to the interfacing system. The impact of the failure of those components on the system being categorized can be assessed and documented but the overall categorization of the component may not be able to be completed until the interfacing system is categorized. In this case, the component would at its original classification until the impact on the interfacing system could be determined.
- In certain instances, the failure of a component in the system being categorized may be postulated to result in a failure of an interfacing system. If the importance of an interfacing system is not known at the time of categorization of the system under consideration, the impact of the failure of the component on the other system may not be able to be determined. An example is the failure of an air-operated solenoid in one system that results in a degraded condition in the air system where the impact of the degradation cannot be determined at the time of categorization of the air-operated valve.

In each of these cases, two options for proceeding with RISC are available:

- Complete the assessment of the impact on the interfacing system, or
- Leave the component categorization unchanged until the categorization of the interfacing system is undertaken.

In the latter case, the component remains at its previous safety level (e.g., a safety-related SSC would remain RISC-1) until the interfacing system categorization is completed. A note would be generated to identify that the component had been identified as a candidate RISC-3 component with respect to the system to which it belongs (per its tag number), but would also have to be categorized as RISC-3 with respect to its impact on the supporting system.

There is typically no systematic process to identify the boundaries between the components in the system being categorized and the support systems that interface with a component. Support systems include electrical power, control power, instrumentation, cooling water and compressed air. This identification must be individually performed for each applicable component to identify the isolation device, if one exists, that prevents a failure in the system being categorized from impacting the support system, and whether that isolation device is part of the system being categorized. If an isolation device can be identified and the isolation device is not part of the system being categorized, then the categorization can still proceed but the importance of the isolation device is recorded for any future categorization of the system to which the isolation device belongs.

Identification of Active System Functions

The system functions from the Maintenance Rule are compared to the Design Basis functions for the WCGS. Differences normally exist between the two sources and are primarily a result of

the different objectives of the Maintenance Rule and Design Basis programs. The Design Basis functions identify the high level requirements for plant operation or safety that a system needs to fulfill. The Maintenance Rule functions are designed to track SSC performance according to the functions that a group of SSCs would perform. All differences between sources are noted and resolved to establish a comprehensive list of active functions that reflect both the Maintenance Rule and the Design Basis functions.

The purpose of defining functions for the 50.69 categorization process is to permit the determination of the safety significant of the function with the understanding that the safety significance of the components that support that function is categorized the same as the function. This simplifies the categorization task in that each component does not have to be examined individually. The subsequent mapping of components to functions ensures that each system component has at least one associated system function. Thus the methodology provides the opportunity to define the appropriate functions for the 50.69 categorization as illustrated in the examples discussed below.

In the assessment of the EN and GK systems, the Maintenance Rule functions were used as the active component functions since these were found to cover all of the Design Basis functions. In one case, Maintenance Rule function GK-04, "Provide conditioned environment for cable spreading rooms, access control, and other areas of Control Building," covered four design basis functions:

- The control building supply system functions to provide conditioned outside air for ventilation and cooling to each level of the control building.
- The access control A/C system functions to provide a suitable environment for personnel comfort.
- The counting room cooling coil and fan coil unit function to provide a suitable environment for personnel and equipment located in the counting room.
- The Security Access Secondary (SAS) room A/C system functions to provide a suitable environment for personnel and equipment located within the SAS room.

The design basis intent was for air from the control building supply system to be supplied to the space above the access control area to remove the heat generated by the Class 1E power cables. This cooling is provided to minimize the amount of cooling required for the spaces below. This function also isolates on receipt of Control Room Ventilation Isolation Signal. The functional pathway is from the outside air intake through the Control Building Supply Air Unit to various heat exchangers and areas throughout the Control and Auxiliary Building. Based on engineering judgment, it was determined that none of the four individual Design Basis functions would be categorized differently if they were considered separately. Therefore, only the single Maintenance Rule function was carried forward in the assessment. If one of the four functions would be determined to be safety significant, then either all four would be categorized as safety significant or they could have been split into the four Design Basis functions and assessed individually. The initial combining of the functions is conservative in terms of the categorization.

In addition to the active functions defined using the process previously described, additional functions can be identified to further facilitate the categorization process where it is known that certain groups of components do not support any of the previously defined Maintenance Rule or Design Basis functions. If it can be shown that the failure of the components mapped to these new functions would not fail or significantly degrade any of the other functions to which they share a functional flow path, then the definition of additional functions simplifies the categorization process to the extent that these components may be considered as a group

rather than individual exceptions to the previously defined active functions. Examples of additional active functions are:

- Provide system vent and drain capability – Vent and drain lines are typically 1 inch lines connected to the major system piping to facilitate maintenance activities.
- Provide alarm and indication – Any alarm or indication that is determined to support a safety significant operator action would not be included in this function. All other components that provide alarm and indication would be included based on the rationale that they simply provide information or status of the plant condition.
- Provide instrumentation isolation – Instrumentation lines are typically small piping less than one half inch in diameter and contain isolation root valves.

These new functions are categorized in the same way as the other functions, as described in succeeding sections.

Mapping of Components to Active Functions

Mapping system components to active functions allows the risk categorization results to be applied to components that would otherwise not have risk information available to properly assess their risk significance. All components are mapped to at least one active function. Components may have more than one active function.

The mapping of system components to active functions is done by using both the component list and P&IDs from WCGS and the active functions developed for the 50.69 implementation as discussed above. The first step in the mapping process is to identify the major pieces of equipment (e.g., pumps, isolation valves, and heat exchangers) required to support each active function. These major pieces of supporting equipment were identified in the Maintenance Rule program, the design basis documentation, or the plant system descriptions. After the major components of each function were identified, the system P&IDs, or other drawings or schematics (e.g., electrical schematics) were then used to physically draw a functional flowpath of the function, from start to finish. This process identifies all components along the supporting flowpath of each function. In some cases, a function only applies to a set of components sharing some common attributes (e.g., system isolation) and therefore would not be associated with a flowpath. All components are then mapped to all functional flowpaths they support as identified on the P&IDs. For some function groupings (e.g., instrument isolation valves), there is no functional flowpath and the components are mapped to the groupings based on the component description or component type label in the database. This does not require association with the P&IDs.

In some cases, component may not appear on the P&IDs because they are either piece parts of a larger component or are not typically included on the drawings. These components are mapped to functions by associating them with other components based on component description or name. For instance, a motor or limit switch for a given motor operated valve will be linked to the same active functions as the motor operator based on the associated valve name or description. The process continues until all components are mapped to at least one active function or grouping.

At this point of the process it is important for every component to be mapped to at least one active function and for each component to be included with all active functions that it supports. The system P&IDs are studied to identify components that have not been mapped. The rationale applied to these components differs based upon the inclusion of each identified component on the system equipment list. Components found on the P&IDs that are not included on the system equipment list are closely scrutinized to decide if these components

belong to the system being analyzed. If the components should be included within system bounds, the component is added. If a component that has not been mapped is included on the system original component list, the active system functions are reviewed and modified if necessary to include the component. If the component cannot be mapped to an identified active function, an active function is added to group components that support a similar function. This process continues until all components are accounted for.

The marked-up P&IDs are included with the IDP package to be used as a tool to facilitate the review of components associated with each function. A listing of the components mapped to each function is also provided for IDP review and approval.

3.3 COMPONENT SAFETY SIGNIFICANCE ASSESSMENT

Section 5 of the NEI 00-04 guidance discusses the use of the plant-specific risk information to identify components that are candidate safety significant to fulfill the requirements of paragraph 50.69(c)(1)(ii). The process includes consideration of the component contribution to full power internal events risk, fire risk, seismic risk and other external hazard risks, as well as shutdown safety.

WCGS Implementation

The WCNOC component safety significance assessment followed the guidance in Section 5 of NEI 00-04, except for the PRA technical adequacy requirements. Per 50.69 (c)(1)(ii), the fulfillment of the PRA technical adequacy requirements will be discussed in a separate submittal in order to gain full NRC approval for implementation of 50.69 at WCGS. The details of the implementation are described in the following paragraphs.

For the WCGS, the following PRA assessments are used in the categorization process:

- The Internal Events At-Power risk assessment uses the quantitative PRA Model,
- The Fire Initiating Event risk assessment uses the qualitative fire risk assessment based on the EPRI FIVE screening tool to define a set of safe shutdown components and pathways,
- The Seismic Initiating Event risk assessment uses the qualitative seismic risk assessment based on the SMA approach to define a set of safe shutdown components and pathways,
- Other External Events risk assessment uses the qualitative external events risk assessment that includes high wind and tornado initiated events to define a set of safe shutdown components and pathways,
- Shutdown Initiated Events risk assessment uses the qualitative shutdown risk assessment based on the identification of safe shutdown pathways from the safe shutdown equipment list.

Internal Events Assessment

The risk assessment of internal initiating events is based upon the WCGS At-Power PRA Model. The 1998 PRA model is used for the current categorization of the study of the EN and GK Systems. This PRA does not meet the PRA technical adequacy requirements described in NEI 00-04. Therefore the results of the categorization of these two systems using the process described in this document will not be implemented until the risk significance of components can be validated using against PRA models that meet the PRA technical adequacy requirements in NEI 00-04. However, the process described in this report for using the PRA results in the categorization process will be applied to any system for which the proposed 50.69 categorization process is being implemented.

WCNOC uses the WinNUPRA PRA model calculator (Reference 7) to evaluate component importance from the PRA model. Components that are modeled in the PRA with failure modes that are above the truncation limit established for the WinNUPRA PRA model calculator are evaluated based on the safety significance criteria established in NEI 00-04. Additional components modeled in the PRA that fall below the truncation limit and components not modeled in the PRA are not evaluated to the internal events assessment criteria since their risk importance associated with the internal events analysis is assumed to be sufficiently low that their risk importance will be certainly be low. The CDF and LERF truncation limit to be used for establishing the risk importance of components and functions for implementation of 50.69 will be 1.0 E-10 per year.

NEI 00-04 provides the guidance for evaluating component importance. The Fussell-Vesely (FV) values for each modeled basic event of a given component are summed together and evaluated against the acceptance criterion of 0.005. If the sum of the basic event FV values is greater than 0.005 for a component, then it is categorized as high safety significant (HSS). The Risk Achievement Worth (RAW) for each modeled basic event of a given component is reviewed to determine the maximum value. If the maximum value of RAW for a basic event for a component is greater than 2, then the component is categorized as HSS. Finally, the maximum common cause failure (CCF) RAW portion of the component failure mode is compared against an acceptance criterion of 20, as discussed in NEI 00-04. If the sum of the FV values for the basic events for a component is less than 0.005, the maximum RAW value for the basic events is less than 2, and the common cause failure RAW value is less than 20, then the component is identified as low risk significant and is then a candidate low safety significant (LSS) component.

This ranking is then applied to all active functions that the component supports. However, not all failure modes for a given component apply to all active functions. Care is taken at this point of the analysis to apply the safety significance of a component failure mode to active functions where the same component failure mode is applicable. The preliminary component mapping is closely reviewed when assigning safety significance. For example, if a valve is required to open to support a function, then only basic events that would result in the valve failing to open should be included in the FV, RAW and CCF RAW assessments.

In cases where the PRA importance measures fall just below the cut-off values for high versus low safety significance, the component is categorized as low safety significant, but the close proximity of the RAW or FV is highlighted for the IDP consideration. This is considered important information for the IDP because one of the overall goals of the categorization process is to assure stability of the categorization results as changes to the PRA model are made to reflect future operating experience and PRA model improvements. One school of thought is that the components whose RAW or FV is just below the acceptance criteria are the most likely to change from low safety significant to high safety significant. By highlighting the close proximity of the RAW and FV to the acceptance criteria to the IDP, an informed, measured decision can be taken.

As specified in NEI 00-04, a series of sensitivity assessments are also performed for any quantitative model used. The purpose of the sensitivities is to identify any significant changes in the PRA results (i.e., component FV or RAW values) when PRA models and assumptions that may have large uncertainties are varied over a reasonable range of values. Should a component exceed the HSS criteria for either FV, RAW or CCF RAW based on one or more sensitivity study then the results are highlighted to the IDP members for their consideration. While all of the sensitivity results are presented to the IDP, this highlighting is meant to focus IDP attention on these components.

The following is a list of the sensitivity studies required by NEI 00-04 for each quantitative model, with some additional studies required based on the findings from the PRA technical adequacy assessment:

- Increase all human error basic events to their 95th percentile value
- Decrease all human error basic events to their 5th percentile value
- Increase all component common cause events to their 95th percentile value
- Decrease all component common cause events to their 5th percentile value
- Set all maintenance unavailability terms to 0.0
- Any applicable sensitivity studies identified in the characterization of PRA adequacy

The human error sensitivities are performed by using the error factor assigned to the human error probabilities (HEP) from the PRA model to determine the 5th and 95th percentile values. This applies to all HEPs in the PRA model and just not those HEPs that may be associated with the system being categorized. Similarly, the 5th and 95th percentile values for common cause failure for components are also derived from the error factors used on the common cause model in the PRA. As in the case of the HEPs, the CCF is changed for all components modeled in the PRA. The maintenance unavailability for components is set to zero for all components modeled in the PRA simultaneously.

Since PRA technical adequacy has not been addressed prior to this initial categorization, no additional sensitivity studies were identified. If additional sensitivity studies are identified during the technical adequacy review of the PRA, they would be included in future the categorization processes.

Fire Risk Assessment

The WCGS fire risk assessment is based upon an evaluation of the risk importance using the FIVE methodology. Following the completion of the technical adequacy assessment for the at-power internal events PRA, a review of the FIVE assessment will be completed to assure that it is consistent with the at-power internal events PRA and represents the current plant design and operation. Therefore, the process for using the FIVE results is described in this section. The results of the categorization for the EN and GK Systems will be re-visited when the quality of the FIVE assessment is completed.

The results of the FIVE assessment are provided in the WCGS IPEEE. Qualitatively, each area or zone of the plant is evaluated to determine its susceptibility to a fire scenario. If a zone is not susceptible, then the zone is screened out of the FIVE analysis and considered low risk. If a zone is susceptible, the zone becomes part of a screened scenario which prompts a quantitative assessment of that particular zone.

Section 5.2 of NEI 00-04 provides the guidance for evaluating fire risk when using the FIVE assessment. If a component participates in an unscreened event it is identified as a candidate safety significant component. If a component participates in a screened event it may also be potentially safety significant unless not crediting the component will not result in its associated scenarios becoming unscreened. The EN and GK Systems were reviewed to determine which components would support these scenarios.

If a quantitative fire PRA model that meets applicable quality standards is developed for WCGS in the future, the fire risk assessment guidance would be similar to that described for the internal initiating events assessment. The results of the fire risk assessment would be combined with other quantitative risk assessment results in an integral assessment to assess the overall risk

impact. Section 5.2 of NEI 00-04 provides guidance for using the quantified fire PRA model. Sensitivity studies similar to those presented above for the internal events would be performed using the fire PRA model, with the addition of a study with no credit for manual suppression.

Seismic Risk Assessment

The WCGS seismic risk assessment was performed using the seismic margins analysis (SMA) approach. The results of the SMA are the Safe Shutdown Equipment List (SSEL) documented in Table 3.6 of the WCGS IPEEE. The SSEL identifies all supporting components required to achieve a safe shutdown as determined by the SMA. Section 5.3 of NEI 00-04 provides the guidance for evaluating the seismic risk when using the SMA. If a component supports the safe shutdown path then it must be considered HSS. The EN and GK Systems were reviewed to determine which components would support these functions.

Following the completion of the technical adequacy assessment for the at-power internal events PRA, a review of the SMA assessment will be completed to assure that it is consistent with the at-power internal events PRA and represents the current plant design and operation. Therefore, the process for using the SMA results is described in this section. The results of the categorization for the EN and GK Systems will be re-visited when the quality of the SMA assessment is completed.

If a quantitative seismic PRA model that meets applicable quality standards is developed for WCGS in the future, the seismic risk assessment guidance would be similar to that described for the internal initiating events assessment. The results of the seismic risk assessment would be combined with other quantitative risk assessment results in an integral assessment to assess the overall risk impact. Section 5.3 of NEI 00-04 provides guidance for using the quantified seismic PRA model. Sensitivity studies similar to those presented above for the internal events would be performed for the seismic PRA model, with the addition of a study using correlated fragilities for all SSCs in an area.

Other External Risks Assessment

The WCGS IPEEE has considered both high winds and flooding as external events that could also affect plant risk. NEI 00-04 states that the safety significance of components related to other external events should be evaluated similar to the qualitative seismic risk assessment by determining whether or not components support the safe shutdown path. The EN and GK Systems were reviewed to determine which components would support these functions.

Following the completion of the technical adequacy assessment for the at-power internal events PRA, a review of the external events assessment will be completed to assure that it is consistent with the at-power internal events PRA and represents the current plant design and operation.

If a quantitative PRA model for other external events that meets applicable quality standards is developed for WCGS in the future, the applicable external events risk assessment guidance would be similar to that described for the internal initiating events assessment. The results of the external events risk assessment would be combined with other quantitative risk assessment results in an integral assessment to assess the overall risk impact. Section 5.4 of NEI 00-04 provides guidance for using the quantified other external events PRA model. Sensitivity studies similar to those presented above for the internal events are to be performed for the other external events PRA model, as well.

Shutdown Risks Assessment

For a qualitative shutdown risk assessment, it is important to identify the key safety functions and the components that support those functions. The key safety functions are; decay heat

removal, inventory control, power availability, reactivity control and containment. The EN and GK Systems were reviewed to determine which components would support these functions. Also, the Safe Shutdown Equipment List (SSEL) from the IPEEE was reviewed to identify the list of equipment required to support safe shutdown (although this is redundant to the seismic risk assessment using the SMA methodology). Components that support any of the key safety functions or are listed on the SSEL are considered safety significant. One final consideration for all other components is whether or not the component could initiate a shutdown accident condition requiring mitigation.

Following the completion of the technical adequacy assessment for the at-power internal events PRA, a review of the shutdown assessment will be completed to assure that it is consistent with the at-power internal events PRA and represents the current plant design and operation.

If a quantitative PRA model for shutdown events that meets applicable quality standards is developed for WCGS in the future, the applicable shutdown risk assessment guidance would be similar to that described for the internal initiating events assessment. The results of the shutdown risk assessment would be combined with other quantitative risk assessment results in an integral assessment to assess the overall risk impact. Section 5.5 of NEI 00-04 provides guidance for using the quantified shutdown PRA model. Sensitivity studies similar to those presented above for the internal events would be performed for the shutdown PRA model.

Integral Assessment

When multiple quantitative risk results are available for a given component, an integral assessment is required to provide an overall assessment of the risk importance. The integral assessment weights the importance from each risk contributor by the fraction of the total core damage frequency of that contributor.

This step is not necessary at present because the internal events PRA model is the only source of quantitative risk insights. However, if quantitative PRA models for fire, seismic, external events or shutdown are developed for WCGS in the future, the integral risk assessment would be performed using the methodology described in the Section 5.6 of NEI 00-04.

3.4 DEFENSE-IN-DEPTH ASSESSMENT

Section 6 of the NEI 00-04 guidance discusses the process to assure that defense in depth is maintained for safety-related components that are found to be low safety significant to fulfill the requirements of paragraph 50.69(c)(1)(iii). The process includes consideration of defense-in-depth related to core damage, large early release and long term containment integrity.

For functions identified as being safety-related and candidate low safety significant after the quantitative and qualitative risk assessment results are considered, defense-in-depth must be considered. If D-I-D is found to be maintained, then the candidate low safety significance may be considered. If defense-in-depth (D-I-D) cannot be shown to be maintained, the function (and its associated components) would be considered potentially high safety significant.

WCGS Implementation

The WCNOC defense in depth assessment followed the guidance, without exception, in Section 6 of NEI 00-04. The process was applied in a straight-forward manner and no additional clarification of the implementation process is required.

In cases where any of the defense in depth considerations was not maintained according to the NEI 00-04 process, the component/function was categorized as candidate high safety significant. If all of the considerations were met (i.e., defense in depth was maintained), then the component or function was identified as candidate low safety significance. In cases where components or functions are identified as safety significant because of defense in depth

considerations, the safety significant attributes are defined by identifying the performance aspects and failure modes of the SSC that contribute to it being safety significant.

3.5 PRELIMINARY ENGINEERING CATEGORIZATION OF FUNCTIONS

Section 7 of the NEI 00-04 guidance discusses the process of integrating the results of the Component Safety Significance Assessment and Defense-In-Depth Assessment tasks to provide a preliminary categorization of the safety significance of system functions to fulfill the requirements of paragraph 50.69(c)(1)(ii).

WCGS Implementation

The WCNOC system engineering assessment followed the guidance, without exception, in Section 7 of NEI 00-04. The details of the implementation are described in the following paragraphs.

Any function/SSC that was determined to be potentially high safety significant from the internal events PRA-based safety significance assessment is assigned HSS. If the function SSC was found to be low safety significant from the internal events PRA, but high safety significant from fire, seismic, external events or shutdown risk assessments, then the results of this assessment, along with the integral PRA assessment results are documented and presented to the IDP for final categorization. Similarly, if the function SSC was found to be low safety significant from the internal events PRA, but the defense in depth or the PRA sensitivity studies identified that the function/SSC is potentially safety significant then the results and their basis are identified to the IDP for their consideration. All other functions/SSCs are preliminarily assigned candidate low safety significance and the basis for that determination is documented.

Once a system function is identified as safety significant, then all components that support this system function are assigned a preliminary safety significant categorization. Due to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC (or part thereof) is assigned the highest risk significance for any function that the SSC or part thereof supports. All preliminary categorization results are then presented to the IDP for review and final categorization.

For safety significant functions/SSCs, the critical attributes that make the function/SSC safety significant are identified and used as input to the treatment redefinition process. Critical attributes are to include high level features of the SSCs that contribute to the safety significance of the function, such as provide flow, isolate flow, etc.

The results of the compilation of risk information and safety significant attributes are documented for the IDP. Figure 7-2 from NEI 00-04 was used for the WCGS assessment.

3.6 RISK SENSITIVITY STUDY

Section 8 of the NEI 00-04 guidance discusses the process of determining the risk sensitivity of the categorization results in terms of changes in risk. This risk sensitivity is an integral assessment and considers the potential impact of the categorization for all systems to which 50.69 is applied. This risk sensitivity fulfills the requirements of paragraph 50.69 (b)(2)(iv) and 50.69(c)(1)(iv).

WCGS Implementation

The WCNOC system engineering assessment followed the guidance, without exception, in Section 8 of NEI 00-04. The details of the implementation are described in the following paragraphs.

The risk sensitivity study was not performed due to the nature of modeling the EN and GK systems in the PRA. The PRA results only identified a select number of components, all of

which met the criteria for HSS. Thus, there was no need to perform the sensitivity study for these systems. However, this section describes the approach that will be used in future system evaluations.

The final step in the process of categorizing SSCs into risk-informed safety classifications would involve the evaluation of the bounding possible change in risk if the unreliability of all RISC-3 SSCs simultaneously increases by a significant amount. Increasing the unreliability of all low safety significant SSCs by a factor of 3 to 5 provides an indication of the bounding change in risk (CDF and LERF) if there were a degradation in the performance of all low safety significant SSCs. The basic events for both random failure events and common cause events would be increased for failure modes of the component relevant to the function being considered. The increase in the common cause failure rate would occur as a result of the increase in the basic event failure rate from which the common cause failure rate is determined.

In identifying the specific factor to be used in the risk sensitivity study, two considerations should be addressed. The first factor is the cumulative risk increase that would be computed if the unreliability of those SSCs were assumed to simultaneously increase by that factor. That is, the factor used can not lead to exceeding the quantitative acceptance guidelines of Reg. Guide 1.174. The second factor is the ability of a monitoring program to detect a change of that factor. This includes consideration of: a) the currently expected number of failures for the number of demands/hours of operation, and b) the expected number of failures for the expected future number of demands/hours of operation for the population of SSCs that are expected to be classified as low safety significant.

WCNOC proposes to use a factor of 3 for the increase in unreliability of RISC-3 SSCs in this assessment. The corrective action requirements of 50.69 (d)(2)(ii) would result in an early determination of degraded performance of a RISC-3 SSC such that an increase in unreliability by a factor of 3 would not be likely to occur for any single RISC-3 component. The simultaneous increase in unreliability for all RISC-3 SSCs is considered to be an incredible event.

When system PRA information is available for a specific system, this sensitivity study would be performed for each individual plant system as the categorization of its functions is provided to the IDP. This is not a requirement of NEI 00-04, but rather would be preformed to provide additional information to the IDP. Thus, a sensitivity study would be performed for the system, and a cumulative sensitivity would be performed for all the SSCs categorized (past and present) using this process. This would provide the IDP with both the overall assessment of the potential risk implications and the relative contribution of each system.

These sensitivity studies will be re-visited when the IDP has completed its final categorization for a given system to assure that the conclusions regarding the potential aggregate impact have not changed.

3.7 IDP REVIEW AND APPROVAL

Section 9 of the NEI 00-04 guidance discusses the Integrated Decision-making Panel for determining the final RISC classification for each of the system functions and/or components. The IDP fulfills the requirements of paragraph 50.69 (c)(2).

WCGS Implementation

The WCNOC IDP followed the guidance, without exception, in Section 9 of NEI 00-04. The details of the implementation are described in the following paragraphs.

The WCGS IDP makes the final determination on SSC categorization. The IDP is responsible for oversight of the categorization process, review and approval of SSC categorization, and

procedure and working practice development. The pilot IDP was directed in accordance with the draft WCNOC 10 CFR 50.69 IDP Duties and Responsibilities procedure. The procedure is currently a draft procedure because the feedback from its use in the pilot IDP activities will be used to finalize the procedure prior to further usage.

IDP Selection

The WCGS IDP serves as a review and approval group for §50.69 Program activities. The panel reports to the WCNOC Manager of Nuclear Engineering. The IDP is composed of members recommended by respective Division Managers and the §50.69 Program Coordinator. The following functional expertise is specified in the WCGS IDP draft procedure:

- Plant Operations (SRO qualified),
- Safety Analysis (Emphasis on plant design),
- System Engineering (the system engineer for the system being reviewed by the IDP),
- Licensing, and
- Probabilistic Risk Assessment (PRA)

According to the IDP procedure, a chairperson is designated as a member of the §50.69 IDP. The chairperson is not required to represent 1 of the 5 required functional areas. In addition, the procedure recommends that a member from the Maintenance Rule functional area be included. For the Pilot IDP, the member representing the Licensing function also had Maintenance Rule experience.

As a result of the IDP session for the EN and GK systems, WCNOC has determined that a member from the Licensing group will not be required for future IDPs. The rationale is that Licensing provides little or no additional input in the categorization of SSCs. Important licensing issues, such as licensing commitments, are already well known by the systems engineer and/or the safety analysis engineer. Also, Licensing is not a required discipline for the IDP in NEI 00-04, Revision 0.

The IDP members are expected to represent the views and opinions of their respective department. Members may be experts in more than one field; however, excessive reliance on any one member's judgment should be avoided. Each member is responsible for reviewing the categorization information relative to their area of responsibility, including:

- Plant Operations – Use of system components in Abnormal and Emergency Operating Procedures, System Operating Procedures, and Severe Accident Management Procedures.
- Safety Analysis – Use of system components in design and/or licensing basis analyses.
- Systems Engineering – System design basis, system functions and system health report. Also, the occurrence of system components in event or inspection reports, licensing commitments and other licensing documentation
- Licensing – Occurrence of system components in event or inspection reports, licensing commitments and other licensing documentation.
- Probabilistic Risk Assessment – The representation of the system and system components in the plant PRA model and relevant risk importance and uncertainties for the system components.

Training of IDP Members

All members of the IDP received training regarding their responsibilities and the §50.69 categorization process. The WCNOC IDP training uses Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program IDP Training Wolf Creek Generating Station.

All members of the IDP will receive refresher training every three years or whenever plant personnel who received the initial training have not served on an active IDP in the past 18 months. Records of training for the IDP are maintained in accordance with WCNOC training requirements.

Members of the WCNOC IDP were trained prior to the IDP sessions for the EN and GK systems. The training focused on the background and rationale needed for each member of the IDP. Introductory discussion of the proposed §50.69 rule included discussion of special treatment exemptions, key requirements, key §50.69 program documents, Risk-Informed Safety Classifications (RISC), categorization guiding principles and an overview of the categorization process. The NEI 00-04 categorization process and the ASME Code Case N-660 categorization processes were discussed to introduce the preliminary categorization completed prior to the IDP. WCGS PRA adequacy and relevant PRA results were presented to provide a summary of the information that is in the PRA model for use in the categorization process. Once the background was discussed, IDP scope, responsibilities, and process were presented. The active component categorization training included RISC-3 considerations for risk analysis, defense-in-depth, and additional considerations for some candidate LSS components. Passive categorization training focused on ASME Code Case N-660 considerations using defense-in-depth for medium and low consequence category SSCs and aging and condition monitoring. Training was concluded with a discussion of the new categorization considerations based on changes in ASME Code Case N-660 and NEI 00-04 resulting from the WCGS IDP from December 2003.

The IDP understood that the PRA information, upon which the categorization results that the IDP was being asked to review and make decisions, would be updated at some future time before implementation of 50.69, as required by the rule. The IDP members also understood that for this unique case, the IDP would be re-convened after: 1) the PRA technical adequacy requirements were assessed according to NEI 00-04, 2) the necessary PRA updates were completed, and 3) the categorization results presented at this session were updated to reflect those PRA updates. Therefore, the categorization and IDP decisions discussed in this report describe and validate a process for categorization to be used by WCNOC to comply with the requirements of the 50.69 rule. The categorization results are of secondary importance since they may change when 50.69 is actually implemented at WCGS, using a PRA that meets the technical adequacy requirements of NEI 00-04 and the 50.69 rule.

IDP Decision-making Process

The §50.69 Program Coordinator provides material to be reviewed by the §50.69 IDP. The material provided to the IDP will meet the WCNOC quality assurance requirements for basic dispositions. Sufficient time is allowed for a comprehensive review by the §50.69 IDP members prior to the meeting. The level of detail to which each member performs technical reviews of the material will be consistent with the member's knowledge and experience. Members are not required to review technical material in detail for areas outside their areas of expertise.

The proposed categorization is presented to the IDP in logical pieces to facilitate the decision-making process. The IDP members discuss the proposed categorization and supply rationale to the process where needed. Each IDP member identifies to the entire IDP any issues related to

the system, system components, or functions from their area of responsibility. Following discussion, the IDP chairman calls for a vote on the proposed categorization.

Approval of any recommendation by the IDP requires concurrence by a majority decision of members present. The IDP Chairman votes only to resolve a tie vote. All dissenting opinions are recorded and reviewed by the plant management. If appropriate, plant management has the authority to suspend the IDP decision and require further consideration.

IDP Records

A recording secretary is present at all IDP sessions to capture the IDP decisions on each function or SSC as well as the basis for the decision and any other relevant IDP discussions. While the entries at the bottom of the form for active and passive component assessments (e.g., Figure 7-2 of NEI 00-04) provide a summary of the IDP decision and discussion, the IDP minutes are intended to provide a more detailed record of the IDP proceedings.

Per the draft WCNOC 10 CFR 50.69 IDP Duties and Responsibilities procedure, the IDP minutes will be reviewed and approved by each IDP member to verify that the minutes accurately reflect the IDP discussions and any insights derived from the discussions. The IDP minutes and the completed active and passive component will be maintained for the life of the plant in the WCGS document control system.

3.8 SSC CATEGORIZATION

Section 10 of the NEI 00-04 guidance discusses the SSC categorization for determining the final RISC classification for each of the system components. This fulfills the requirements of paragraph 50.69 (c)(2).

WCGS Implementation

The WCNOC SSC categorization followed the guidance, without exception, in Section 10 of NEI 00-04. The details of the implementation are described in the following paragraphs.

Once the preliminary engineering categorization of functions is reviewed by the IDP and the final classification is assigned to each function by the IDP, the final safety significance (either high or low safety significance) is assigned to each active function, as well as all supporting components.

If a component supports a HSS function, then the component will be designated as high safety significant (i.e., either RISC-1 or RISC-2 as appropriate). If the component only supports low safety significant functions, then the component may be classified as low safety significant (i.e., either RISC-3 or RISC-4 as appropriate). Thus, if at least one function supported by a component is ranked high safety significant, then the SSC will also be ranked high safety significant, regardless of how many other low safety significant functions it may support.

In the future, WCNOC may perform a more detailed mapping and categorization process to identify SSC ranked high safety significant that may not actually support the critical attributes of the high safety significant function. In this case, the RISC process described above will need to be repeated and another IDP would be convened to review and approve the modified results.

Risk-Informed Safety Classification - 50.69 Summary Report

Plant Name - Unit No. *SYSTEM NAME* **S1**

ACTIVE ASSESSMENT SUMMARY

FUNCTION ID **S1-01**

FUNCTION DESCRIPTION Sample active function #1 for sample system S1.

See attached sheet for associated components.

ACTIVE RISK ASSESSMENT

Risk Assessment Elements	*Rank*	*Comment*
INTERNAL EVENTS ANALYSIS	HSS/LSS	Basis for internal PRA ranking.
FIRE ANALYSIS	HSS/LSS	Basis for fire rank.
SEISMIC ANALYSIS	HSS/LSS	Basis for seismic rank.
FLOODING ANALYSIS	HSS/LSS	Basis for other external events rank.
SHUTDOWN ANALYSIS	HSS/LSS	Basis for shutdown rank.
INTEGRATED RISK ASSESSMENT	HSS/LSS	Basis for integral assessment.
RISK ASSESSMENT RANKING	**HSS/LSS**	

Additional Risk Assessment Information

PRA SENSITIVITY ANALYSIS	HSS/LSS	Basis for PRA sensitivity rank.
DEFENSE IN DEPTH ANALYSIS	HSS/LSS	Basis for defense in depth rank.

IDP ASSESSMENT

RISK RANKING	*RISK RANKING BASIS*
HSS/LSS	Basis for the final risk ranking of Active Function ID S1-01.

CRITICAL ATTRIBUTES	*COMMENTS*
If risk ranking is HSS, IDP identifies the critical attributes. If risk ranking is LSS, then no entry is required.	Basis for the HSS critical attributes.

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Figure 3-2: Active Assessment Summary Report, Page 1

Risk-Informed Safety Classification - 50.69 Summary Report

Plant Name - Unit No. *SYSTEM NAME* **S1**

ASSOCIATED COMPONENTS

SAFETY CLASS	*COMPONENT ID*	*COMPONENT DESCRIPTION*
SR	S1PUMP-A	Description of S1PUMP-A
SR	S1PUMP-B	Description of S1PUMP-B

Date Page X of Y

Figure 3-3: Active Assessment Summary Report, Page 2

4. CATEGORIZATION OF PASSIVE FUNCTIONS

ASME Code Case N-660 was approved by the ASME Board on Nuclear Codes and Standards in 2002 and issued for trial use. Through its trial use, a number of modifications have been proposed, primarily as a result of trial usage at the December 2003 WCGS IDP. These changes were reflected in the draft revision to Code Case N-660 that was used in the April 2004 IDP at WCGS. Significant changes from the approved version as discussed in Appendix A. All discussion of Code Case N-660 in Section 4 of this report, unless otherwise noted, will refer to Revision 0 as approved by the ASME and endorsed by the NRC.

Note that Appendix A was originally included to show the differences between the categorization guidance used in the WCGS IDP and that which was approved by ASME and endorsed by NRC. In the course of the NRC review of this topical report, the information in Appendix A for passive categorization has been revised to reflect a proposed passive categorization process for future implementation of 50.69 at WCGS. In all cases, if there are inconsistencies between the text in this section and the guidance in Appendix A, the guidance in Appendix A should be considered to supersede the following text.

4.1 SCOPE IDENTIFICATION

Code Case N-660 paragraph I-2.0 requires the definition of the boundaries to be included in the scope of the RISC evaluation and also defines the requirements for Class 1 items.

WCGS Implementation

The WCNOC SSC scope identification followed the guidance, without exception, in Paragraph I-2.0 of Code Case N-660. There was no Class 1 piping in either the EN or the GK systems. Consistent with the NRC endorsement of Code Case N-660 in Regulatory Guide 1.147, Revision 14, WCNOC will classify all Class 1 piping as high safety significant, unless changes are endorsed by NRC in further revisions to Regulatory Guide 1.147.

4.2 COMPONENT MAPPING

Code Case N-660 paragraph I-3.0 requires that all pressure retaining items and their supports be evaluated by defining piping segments that are grouped based on common conditional consequences.

WCGS Implementation

The WCNOC SSC component mapping followed the guidance, without exception, in Paragraph I-3.0 of Code Case N-660.

The WCGS RI-ISI program was used as a starting point for identifying the appropriate grouped piping segments for the pressure boundary component assessment, as defined in ASME Code Case N-660. The WCGS RI-ISIS program is a controlled program at WCGS and uses the "EPRI RI-ISI methodology." WCNOC implemented the RI-ISI program for Class 1 and 2 piping, including the Class 2 piping in the EN System. Information for these passive piping segments was gathered and used for the Code Case N-660 passive assessment. There was no Class 1 or 2 piping in the GK System, so no RI-ISI information was available for use in the passive categorization of the GK system.

Passive piping segments are runs of piping whose failure would result in nearly identical consequences, including consideration of spatial effects. Room boundaries (walls) are one means used in the RI-ISI program to establish the boundaries of piping segments based on potential differences in spatial effects. Pumps also establish boundaries of piping segments

since the system operating conditions (pressure, flow velocity, etc.) would be different on the suction and discharge sides of a pump. Passive piping segments were identified for the system boundaries established by the system equipment list and checked by the active component mapping.

For those piping segments that had not been originally identified in the WCGS RI-ISI program, additional passive piping segments were created based on piping segments having similar failure consequences. The failure consequences are used later in the analysis as part of the passive risk ranking process to determine the consequence categories.

All components with a pressure boundary function were mapped to one of the passive piping segments based on the information gathered for the active component mapping discussed in Section 3 of this report. All components with a pressure boundary function were mapped to one and only one pipe segment with the exception of those components that are at the boundary between two pipe segments. In this case, the component is uniquely associated with each pipe segment and is assigned the highest safety significance of the two pipe segments. For example, if one pipe segment was classified as high safety significant and the adjacent pipe segment is classified as low safety significant, the component that is the boundary between the two segments would be categorized as high safety significant.

4.3 REQUIRED DISCIPLINES

Paragraph 1320 of Code Case N-660 requires that the following disciplines be involved in the classification process:

- Probabilistic Risk Assessment
- Plant Operations
- System Design
- Safety or Accident Analysis

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

WCGS Implementation

The WCNOC classification process followed the guidance, without exception, in Paragraph 1320 of Code Case N-660.

The WCGS pilot effort satisfied this requirement by having the 50.69 IDP participate in the passive assessment for the EN and GK Systems. Although Code Case N-660 does not require the use of an IDP, the 50.69 rule does require the use of one when implementing N-660 with NEI 00-04.

Figure 4-1 illustrates the content and format of the information used for the IDP review of the WCGS passive assessment. This form was completed for each pipe segment. The form summarizes the results and insights that were generated in the categorization process and identifies the key information that should be communicated to the IDP for use in the decision-making process.

4.4 CONSEQUENCE EVALUATION

Paragraph I-3.1 of The Code Case N-660 specifies that the potential failure modes for each system or piping segment shall be identified, and their effects shall be evaluated. The results of the failure modes evaluation for each piping system, or portion thereof, shall be classified into one of three impact groups: piping failures that cause an initiating event, those that disable a

system without causing an initiating event, or those that cause an initiating event and disable a system. The consequence category assignment for each piping segment within each impact group shall be selected in accordance with guidance in the Code Case. In assessing the appropriate consequence category, available risk information for all initiating events, including fire and seismic should be considered.

WCGS Implementation

The WCNOC consequence evaluation followed the guidance in Paragraph I-3.1 of Code Case N-660 with the following exception/clarification:

- Consideration of fire, seismic, external events and shutdown was also considered in the evaluation.

A discussion of the consequence assessment, including the exceptions to the Code Case, is discussed in the following paragraphs.

For the Wolf Creek assessment, the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) values for those pipe segments previously categorized as part of the RI-ISI program were used as input to determine the consequence ranking. Table I-5 of Code Case N-660 shows the ranges of HIGH, MEDIUM, LOW, and NONE consequence categories, based on CCDP. For CLERP, the values in Table I-5 of Code Case N-660 are lowered one order of magnitude. The higher of the CCDP or CLERP ranking category for any given segment was assigned to the segment. For example if the CCDP for a given segment was MEDIUM but the CLERP was LOW, the segment was ultimately assigned a MEDIUM consequence category.

For piping segments that were not part of the WCNOC RI-ISI program and therefore did not have a consequence assessment, the consequence assessment was performed by using Tables I-1 and I-2 of Code Case N-660. In using Table I-2 the exposure time to challenge was taken from plant Technical Specification Allowable Outage Times, when the Tech Specs applied to a component in a pipe segment. Otherwise, an exposure time of "all year" was used.

The RI-ISI consequence assessment only included the CCDP and CLERP from the at-power internal initiating events PRA. Also, only qualitative risk assessments exist for fire, seismic, external events and shutdown at WCGS. Therefore, to capture the risk importance of piping segments from the fire, seismic, external events and shutdown qualitative risk assessments, any piping segment supporting a safe shutdown pathway would be a classified as a high safety significant pipe segment. This is consistent with the active component classification process where active SSCs that support safe shutdown pathways are automatically classified as high safety significant and therefore not eligible to be ranked lower by the IDP.

4.5 CLASSIFICATION CONSIDERATIONS

Paragraph I-3.2 of the Code Case N-660 specifies that the Risk Informed Safety Classification is determined by considering the consequence category, in conjunction with other relevant information. Piping segments determined to be a Medium, Low, or None (no change to base case) consequence category shall be determined HSS or LSS by considering the other relevant information for determining classification. A set of conditions, as detailed in the Code Case shall be evaluated and answered TRUE or FALSE. If any of the conditions are answered FALSE, then HSS shall be assigned. Otherwise, LSS may be assigned.

If LSS has been assigned, then the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data.

WCGS Implementation

The WCNOC consequence evaluation followed the guidance in Paragraph I-3.2 of Code Case N-660 with the following exceptions and clarifications:

- A small break was assumed for some pipe segments based on the low potential for a large break because it carries low pressure and low temperature fluid,
- Operator actions were credited for isolation of a pipe segments if clear indication of the pipe break is evident in the control room, procedures for isolation of the pipe break are available, and ample time exists for the operators to diagnose and isolate the break,
- Consideration of fire, seismic, external events and shutdown was also considered in the evaluation, and

- Some of the considerations have been clarified and/or changed as a result of the trial usage at the December 2003 IDP, as discussed in Appendix A of this report.

The classification considerations as described in ASME Code Case N-660 are used in the WCGS analysis for further classification of segments with a consequence category other than HIGH. If the consequence category as described in the previous section is HIGH, this section does not apply and the piping segment is assigned a high safety significant (HSS) overall ranking. For piping segments with a MEDIUM, LOW, or NONE consequence category, HSS or LSS is determined by the following eleven additional considerations from Code Case N-660. All of the considerations in Section I-3.1.3 of Code Case N-660 were changed so that the response (i.e., true vs. false) for passive categorization would match the response for the equivalent consideration from the active categorization in NEI 00-04. The following statements should be answered either TRUE or FALSE with appropriate justification:

1. Even when taking credit for plant features and operator actions, failure of the piping segment will not directly fail another high safety significant function.

   The intent of this statement is to provide another check that all direct failure consequences have been identified as the result of a pipe segment's failure. The purpose of the consequence evaluation, as described in the previous section, is to determine all failure consequences of a pipe segment's failure. Plant system functions, other than those solely within the system being evaluated, have already been considered and evaluated. Therefore, this only serves as a final verification that the consequence evaluation considered all impacts on other systems.

   Piping segments shall be ranked based on the highest safety significance of all of the functions that it supports. If the importance of all functions that it supports has not been completed, the piping segment must retain its original classification until the importance of all supporting systems has also been evaluated.

   The safety significance of the pressure boundary function of a piping segment should be consistent with the safety significance of the active function of a piping segment, except in certain circumstances. All such circumstances would need to be justified on a case by case basis.

   Credit was also taken, in some instances, for the detection of leaks and isolation of breaks to prevent the loss of other high safety significant functions. For credit to be taken, alarm functions must be available to alert the operators to the condition and the need to take action to isolate the leak prior to rendering another high safety significant inoperable.

2. Failure of the piping segment will not result in failure of another high safety significant function (e.g., through indirect effects).

   The intent of this statement is to consider indirect effects such as pipe whip, jet impingement spray and flooding that could result in the failure of another high safety significant function. For piping segments classified in the risk informed ISI program, the indirect effects assessment from that program, which included a system walk-down, was used as a basis for this assessment. For any systems or pipe segment not classified in a risk informed ISI program, a walk-down should be performed to identify any indirect effects. Information from the systems engineer and a senior reactor operator may be substituted for a plant walk-down. In assessing possible indirect effects, credit may be taken for protective barriers and plant layout to mitigate the effects of possible spatial interactions.

3. Failure of the piping segment will not prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions.

The intent of this statement is to consider the plant's ability to reach or maintain safe shutdown conditions given failure of the piping segment. Components that support a primary or alternate safe shutdown pathway for WCGS were considered high safety significant.

4. The piping segment is not relied upon to support an active function in the plant Emergency Operating Procedures, the Abnormal Operating Procedures ,the Severe Accident Management Guidelines or similar guidance as the sole means for successful performance of operator actions required to mitigate an accident or transient, or for achieving actions for long term containment integrity, monitoring of post accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions.

   The WCGS Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) were reviewed to determine the significant mitigating and diagnosis functions. The significant functions from the EOPs and SAMG are: sub-criticality, core cooling, heat sink, RCS integrity and RCS inventory, and containment integrity. Piping segments were evaluated to determine if their failure would result in the failure to diagnose or mitigate one of these functions for the system being categorized. If it was the sole means of diagnosing the function, then the pipe segment would be classified as high safety significant; otherwise a low safety significant classification would be assigned based on this consideration.

5. The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow assisted corrosion program).

   The purpose of this consideration is to assure that any pipe segment for which there is a known active degradation mechanism, but no condition monitoring program, would not be classified as low safety significant. The known active degradation mechanisms and condition monitoring for the system were identified by the system engineer. For the EN and GK systems, no pipe segments were identified as susceptible to a known active degradation mechanism for which a condition monitoring program was not already in place.

6. Failure of the piping segment will not result in releases of radioactive material that would result in the declaration of a general emergency condition.

   The intent of this statement is to consider the radioactive level of the fluid that is contained in the system piping and the location of the system piping, both during normal plant operation and also following a core damage accident.

   None of the piping in the GK system would contain any significant levels of radioactive material during normal operation or after a core damage accident. For the containment spray system, some of the piping segments could contain high levels of radioactivity following a core damage accident as the highly radioactive containment sump water is circulated through the piping segments that perform the spray recirculation function. However, the radionuclides in the containment sump water are not a significant threat to become airborne (i.e., very low volatility and vapor pressures) and they would be contained within the WCGS auxiliary building area where the containment spray recirculation piping is located. Thus, a general emergency condition based on the Environment Protection Agency's (EPA) Protective Action Guides (PAGs) (Reference 8) would not be initiated.

7. A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

No changes are being made to the plant design basis as a result of this program. The implementation of 50.69 simply changes the level of assurance required to maintain component functionality. No components are being removed and all components must maintain function in accordance with the revised treatment program. Therefore, reasonable balance will still be preserved among the prevention of core damage, prevention of containment failure, and consequence mitigation. In spite of the generic nature of the response to this consideration, there was IDP discussion concerning whether the balance had been changed by classifying containment spray system pressure boundary components as LSS. The conclusion of those discussions was that the balance was still maintained based on the 50.69 requirement for maintaining functional capability of the system.

8. Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

   No changes are being made to the plant design basis as a result of this program. No components are being removed and all components must maintain their design basis function in accordance with the revised treatment program. In addition, the changes permitted by the categorization due not increase reliance on any programmatic activities. Therefore, no plant design weaknesses are exposed because of an over-reliance on programmatic activities. Also, since low safety significant pressure boundary components would remain within the scope of condition monitoring programs, there is no over-reliance of programmatic activities.

9. 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).

   No changes are being made to the plant design basis as a result of this program. The implementation of 50.69 simply changes the level of assurance required to maintain component functionality. No components are being removed and all components must maintain function in accordance with the revised treatment program. Therefore, system redundancy, independence, and diversity will be preserved. Any adverse changes to the reliability of redundant, independent of diverse components would have only a negligible impact the plant risk (per the PRA assessment for the component) and would be detected by the monitoring of low safety significant pressure boundary components.

10. Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed (e.g., biofouling).

    No changes are being made to the plant design basis as a result of this program. The implementation of 50.69 simply changes the level of assurance required to maintain component functionality. No components are being removed and all components must maintain function in accordance with the revised treatment program. Defenses against common cause failures will be preserved.

11. Independence of fission-product barriers is not degraded.

    No changes are being made to the plant design basis as a result of this program. The implementation of 50.69 simply changes the level of assurance required to maintain component functionality. No components are being removed and all components must maintain function in accordance with the revised treatment program. The consequence of a failure of any pressure boundary components that may serve as a fission product boundary has already been considered in the initial phase of the classification; any pressure boundary component whose failure would result in significant fission product releases would already be classified as high safety significant. Therefore, independence of fission product barriers

will not be degraded. For example the containment isolation valves inside and outside containment on the containment spray system was classified as high safety significant in the initial phase of the classification based on the consequences of a pressure boundary failure.

If any of the above 11 considerations were FALSE for a given piping segment, HSS was assigned.

The version of Code Case N-660 that was used in the WCGS provided a final consideration that could be used to over-ride the eleven considerations discussed above. This final consideration was that if historical data shows that these failure modes are unlikely to occur and such failure modes can be detected in a timely fashion, the component could be classified as low safety significant. This consideration was not used in the WCGS IDP as a basis for categorizing a pressure boundary component as low safety significant. WCNOC will not use this consideration as part of their 50.69 categorization process (present or future) to over-ride any of the primary considerations discussed above.

4.6 SAFETY MARGINS

Paragraph I-3.2.1 (c) of Code Case N-660 requires that the categorization process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data for all piping segments determined to be low safety significant based on the previously discussed criteria and considerations.

WCGS Implementation

The WCNOC consequence evaluation followed the guidance in Paragraph I-3.2.1 (c) of Code Case N-660 without exception.

For the WCGS analysis, existing safety margins for technical and functional requirements will remain because the only requirements that are relaxed for low safety significant piping and components are those related to treatment. There are no changes made to the plant design basis, licensing basis or safety analysis. Also, the individual PRA sensitivity studies, as well as the overall risk sensitivity study, provide reasonable assurance that the proposed revisions in treatment requirements account for analysis and data uncertainty.

Risk-Informed Safety Classification - 50.69 Summary Report

Plant Name - Unit No. *SYSTEM NAME* **S1**

PASSIVE ASSESSMENT SUMMARY

SEGMENT ID **S1-001**

SEGMENT DESCRIPTION Description of R/R segment S1-001

PASSIVE RISK ASSESSMENT

Risk Assessment Elements	*Result*	*Comment*
CONSEQUENCE RANKING	HSS/LSS	Basis for consequence ranking.
HSS FUNCTION	HSS/LSS/NA	Basis for HSS function question.
INDIRECT EFFECTS	HSS/LSS/NA	Basis for indirect effects question.
SAFE SHUTDOWN	HSS/LSS/NA	Basis for safe shutdown question.
EOPs OR SAMGs	HSS/LSS/NA	Basis for EOPS SAMG question.
INADVERTANT RELEASE	HSS/LSS/NA	Basis for radioactive release question.
REASONABLE BALANCE	HSS/LSS/NA	Basis for reasonable balance question.
PROGRAMMATIC ACTIVITIES	HSS/LSS/NA	Basis for programmatic activities question.
SYSTEM REDUNDANCY	HSS/LSS/NA	Basis for redundancy question.
COMMON CAUSE FAILURE	HSS/LSS/NA	Basis for common cause failure question.
FISSION PRODUCT BARRIER	HSS/LSS/NA	Basis for fission product barrier question.
SAFETY MARGINS	HSS/LSS/NA	Basis for safety margins question.
PRELIMINARY OVERALL RANK	S/LSS/I	Basis for overall risk ranking.

PASSIVE IDP ASSESSMENT

RISK RANKING	*RISK RANKING BASIS*
HSS/LSS	Basis for the final risk ranking of R/R Segment ID S1-001.

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Figure 4-1: Passive Assessment Summary Report, Page 1

Risk-Informed Safety Classification - 50.69 Summary Report

Plant Name - Unit No. *SYSTEM NAME* **S1**

ASSOCIATED COMPONENTS

SAFETY CLASS	*COMPONENT ID*	*COMPONENT DESCRIPTION*
NS	S1MOV-1234	Description of S1MOV-1234
SR	S1PUMP-A	Description of S1PUMP-A

Date Page X of Y

Figure 4-2: Passive Assessment Summary Report, Page 2

5. RISK CATEGORIZATION BASIS

WCGS has a quantitative PRA that models internally initiated events from an at-power condition. The current at-power internal events PRA is known as Revision 4 and was completed in 1998. Other important risk contributors, such as seismic risk, fire risk, other external event risks (high winds, tornadoes, etc.) during power operation, and risk during shutdown conditions have been assessed using qualitative methods that were acceptable at the time the WCGS IPEEE was performed. These were completed in the mid-1990's timeframe.

The PRA input used for the RISC of the components in the EN and GK systems has not been updated to reflect the findings of the PRA peer review nor other quality assessments as described in Regulatory Guide 1.200. Thus, a separate submittal related to the PRA technical adequacy will be made by WCNOC, when the decision is made to implement 50.69, to satisfy the PRA technical adequacy requirements of 50.69. However, the input from the 1998 PRA model is adequate to define the categorization process to be used at WCGS. Following the completion of any PRA modifications that are required to meet the PRA technical adequacy requirements for use in the proposed 10 CFR 50.69 categorization, the present RISC results, which reflect the risk insights from the current PRA, will be updated as appropriate to assure that the results of the categorization process reflect the updated PRA.

The results of the at-power internal events PRA are summarized in this section to show that the results are not significantly different from those reported for other similar plants. Thus, there is reasonable confidence that the process used to categorize SSCs for the EN and GK systems at WCGS will not be affected by changes in the PRA once the acceptable level of PRA adequacy is achieved.

5.1 PLANT-SPECIFIC RISK INFORMATION

The results of the quantification of the PRA model show a CDF value of 5.5E-05/year and a LERF value of 8.3E-07/year.

The initiating event contribution to CDF is dominated by the station blackout event as shown in Table 5-1.

The initiating event contribution to LERF is dominated by the containment bypass events initiated by either a steam generator tube rupture accident or an interfacing system LOCA event as shown in Table 5-2.

The results of the 1998 PRA, in terms of the CDF and LERF, the dominant initiating events and core damage sequences are not significantly different than other industry PRAs.

Table 5-1: Initiating Event Contribution to Core Damage	
Initiating Event	**CDF Contribution**
Station Blackout	48%
Large LOCA	9%
Medium LOCA	8%
Loss of RCP Seal Cooling – At Least One CCW Train Available	7%
Small LOCA	4%
Loss of RCP Seal Cooling Following a Transient Initiator	4%
Loss of All Service Water	4%
Loss of Vital DC Bus NK01	3%
Steam Generator Tube Rupture	3%
Anticipated Transient Without Scram	2%

Table 5-2: Contributors to LERF	
Containment Failure Mode	**CDF Contribution**
Bypass – Steam Generator Tube Rupture	55%
Bypass – Interfacing Systems LOCA	43%
Containment Failure	2%

6. PROGRAM DOCUMENTATION AND CHANGE CONTROL

6.1 DOCUMENTATION OF CATEGORIZATION PROCESS

50.69 (f)(1) requires that the licensee or applicant shall document the basis for its categorization of any SSC under 50.69 (c) before removing any requirements under 50.69 (b)(1) for those SSCs.

WCGS Implementation

The WCNOC documentation process complies with 50.69 (f)(1) without exception for both the active and the passive categorization.

The documentation on the 50.69 categorization process and the list of SSCs that have been subject to the categorization process will be stored in a readily retrievable form for use by WCNOC. A Microsoft Access database has been developed to capture all of the categorization results as well as IDP meeting notes. Each component in a categorized system will be identified as either RISC-1, 2, 3, or 4, based on its safety class and safety significance as determined by the IDP. The documentation will be retained for the life of the facility per the requirements in the WCGS 50.69 IDP procedure.

6.2 CHANGE CONTROL PROVISIONS

Following implementation of 50.69, the paragraph (f)(2) requires that licensees shall update their final safety analysis report (FSAR) to reflect which systems have been categorized, in accordance with 50.71 (e).

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

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

WCGS Implementation

The WCNOC implementation process complies with 50.69 (f)(2) through (f)(4) without exception for both the active and the passive categorization.

In general, the implementation of 10 CFR 50.69 can be divided into two phases: 1) the initial implementation that includes the categorization of SSCs and the application of treatment based on that categorization; and 2) the control of changes to the plant that may impact those SSCs or their categorization basis following the initial implementation. This section discusses how the requirements of 10 CFR 50.69(f) are met for these two phases. For the purposes of this guidance, initial implementation refers to the first application of the 10 CFR 50.69 rule to a particular system. This may be at the time the first system(s) are categorized under 10 CFR 50.69 or it may be at later time if the licensee chooses a phased approach to categorization wherein only a few systems are categorized each year, for several years.

Initial Implementation

Following NRC approval to implement 10 CFR 50.69, any changes to the FSAR that reflect alternative treatment of categorized systems will be captured in the WCNOC FSAR update process. Changes to the FSAR associated with initial implementation for an SSC need not include a supporting review or evaluation under 10 CFR 50.59, but rather a direct reference to the 50.69 categorization process may be substituted since it fulfills the intent of a 50.59 review.

Initial implementation will also entail changes to the licensee's quality assurance plan to reflect alternative treatment for categorized systems. Changes to the quality assurance plan associated with initial implementation for an SSC need not include a supporting review under 10 CFR 50.54(a), but rather a direct reference to the 50.69 categorization process may be used. In addition, any regulatory commitments associated with the special treatment requirements in 10 CFR 50.69(b)(1) for SSCs categorized as RISC-3 are no longer applicable to these SSCs and may be dropped at WCNOC's discretion. However, WCNOC will ensure that any design basis commitments continue to be maintained as defined by 10 CFR 50.2, NRC Regulatory Guide 1.186, and NEI 97-04, Rev. 1 (Reference 9).

The waiver of supporting reviews under 10 CFR 50.59 and 10 CFR 50.54(a) is only applicable to the initial implementation of 10 CFR 50.69 for a system, i.e., for changes in treatment to SSCs based on the results of the categorization process. Any other changes to these SSCs are subject to the applicable change control requirements through the application of NEI 99-04, Revision 1, "Guidelines for Managing NRC Commitment Changes" (Reference 10).

Following Initial Implementation

Subsequent to initial implementation, any changes to alternative treatment for categorized SSCs are subject to applicable change control requirements, e.g., 10 CFR 50.59 and 10 CFR 50.54(a), and will continue to meet the alternative treatment requirements in 10 CFR 50.69.

Changes to categorized SSCs not associated with treatment continue to be governed by the same applicable change control requirements. For RISC-1 and RISC-2 SSCs that have safety significant beyond design bases functions, the WCNOC will maintain reasonable assurance that these functions will be satisfied following the change.

7. PERIODIC UPDATES

7.1 UPDATES BASED ON PLANT DESIGN AND OPERATION

10 CFR 50.69 (e) requires that the licensee shall review changes to the plant, operational practices, applicable industry operational experience, and, as appropriate, update the PRA and SSC categorization. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.

WCGS Implementation

The WCNOC update for plant design and operation complies with 50.69 (e) without exception for both the active and the passive components categorized under 50.69.

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

Scheduled periodic reviews will be performed in accordance with Regulatory Guide 1.200 and will evaluate new insights resulting from available risk information (e.g., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This review will include:

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

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

In most cases, the categorization is expected to be unaffected by changes in the plant-specific risk information. However, in some instances, an updated PRA model could result in new RAW and F-V importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. In these cases, the assessment of whether a change in categorization is appropriate will be based on the changes in absolute value of the importance measures and the changes in risk as described in Table 12-1 of NEI 00-04.

7.2 MONITORING OF RISC-1 AND RISC-2 SSCs

The 50.69 rule, at paragraph (e)(2) requires that the licensee shall monitor the performance of RISC-1 and RISC-2 SSCs, and make adjustments as necessary to either the categorization or treatment processes so that the categorization process and results are maintained valid.

WCGS Implementation

The WCNOC implementation process complies with 50.69 (e)(2) without exception.

RISC-1 and RISC-2 SSCs will be monitored in the same manner as they are presently monitored under 10 CFR 50.65, the Maintenance Rule, with the following clarifications:

1. The monitoring will address all functional failures, not just maintenance preventable functional failures.
2. The scoping requirements of the maintenance rule are expected to envelop practically all RISC-1 and RISC-2 SSCs. However, to the extent that any of these SSCs are not in the maintenance rule scope, appropriate monitoring requirements will be developed for those SSCs.

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

WCNOC will submit a licensee event report under § 50.73(b) for any event or condition that prevented, or would have prevented, a RISC–1 or RISC–2 SSC from performing a safety significant function.

7.3 MONITORING OF RISC-3 SSCs

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

WCGS Approach

The WCNOC implementation process complies with 50.69 (e)(3) without exception.

Performance monitoring of RISC-3 SSCs, as required by 10 CFR 50.69(e)(3), will be established to provide assurance that potential increases in failure rates will be detected and addressed before reaching the rate assumed in the above sensitivity study. WCNOC has not developed plant specific methods for monitoring of correction actions to address degradation of RISC-3 SSCs. Note that WCNOC uses the term RISC-3 SSC to encompass low safety significant safety related active components classified using the NEI 00-04 guidance as well as low safety significant pressure boundary components classified using the passive categorization process described in WCAP-16308-NP. Plant specific methods, when developed in detail, will follow the approach of NEI 00-04.

Specifically, the monitoring program for RISC-3 SSCs, including low safety significant pressure boundary components from the passive categorization process described in this report, will include the flowing features:

- Failures of RISC-3 SSCs will be identified and tracked in a corrective action program.

- Failures of RISC-3 SSCs will be reviewed, as part of the corrective action program, to determine the extent of condition (i.e., whether this failure is indicative of a potential common cause failure).
- Non-failures, such as known active degradation processes, will also be tracked as part of the corrective action program to determine the extent of condition (i.e., the degree of degradation versus the condition monitoring acceptance criterion) and need for corrective actions.
- Failures of RISC-3 SSCs will be assessed for groups of like component types (e.g., motor operated valves, air operated valves, motor-driven pumps, etc.) for the purposes of assessing data from the corrective action program.
- A periodic review of all failures of RISC-3 SSCs, also considering previous component performance history, will be undertaken at least once every two fuel cycles (per the periodic review schedule recommended in Section 12.1 of NEI-00-04) to:
  - Ensure that the failure rate of RISC-3 SSCs in a given time period has not unacceptably increased due to the changes in treatment. The periodic review will validate that the rate of RISC-3 SSC equipment failures has not increased by a factor greater than that used in the integrated risk sensitivity study.
  - Detect the occurrence of potential inter-system common cause failures, and to allow timely corrective action if necessary.

  If the number of failures for a group of SSCs exceeds the expected number of failures by a factor of two or more, a potential adverse trend is identified requiring further assessment. As a result of the assessment, either:
  - The categorization will be revised to reflect the increased failure rates and the ranking of appropriate SSCs will be reviewed, or
  - A corrective action plan will be developed to return the reliability of the SSCs to a level consistent with the categorization.

8. APPLICATION OF RISC-3 TREATMENT REQUIREMENTS

10 CFR 50.69 requires that licensees shall ensure, with reasonable confidence, that RISC–3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC–3 SSCs must be consistent with the categorization process.

WCGS Implementation

RISC-3 SSCs will be exempt from special treatment requirements as described in §50.69 (b)(1). In lieu of those special treatment requirements, other processes are required to ensure, with reasonable confidence, that RISC–3 SSCs remain capable of performing their safety-related functions under design basis conditions as required by §50.69 (d)(2). WCNOC has not developed plant specific methods for inspection, testing and corrective actions for RISC-3 SSCs to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Note that WCNOC uses the term RISC-3 SSC to encompass low safety significant safety related active components classified using the NEI 00-04 guidance as well as low safety significant pressure boundary components classified using the passive categorization process described WCAP-16308-NP.

10 CFR 50.69(d)(2) requires that two elements of safety be maintained:

- Reasonable confidence be maintained that RISC-3 SSCs can perform their design basis functions under their design basis accident conditions, including seismic and environmental conditions and effects throughout their service life, and
- The basis for the categorization of RISC-3 SSCs be validated through monitoring of the performance of RISC-3 SSCs and corrective actions be implemented when the categorization basis is not maintained.

10 CFR 50.69(e) requires that changes to the plant, operational practices, applicable plant and industry operational experience be periodically reviewed and, as appropriate, the PRA and SSC categorization and treatment processes be updated.

To comply with the requirements of 50.69(d)(2), and (e), WCNOC will:

- Procure RISC-3 SSCs in a manner consistent with current practices for commercial grade equipment that includes, as a minimum: a) development of procurement specifications that ensure that the component can perform its design basis function under the appropriate design basis conditions, including seismic and environmental conditions and effects throughout their service life, and b) inspect the equipment upon receipt at the plant to ensure that the proper component was received.
- Periodically maintain and test RISC-3 SSCs in a manner consistent with current practices for commercial grade equipment that includes, as a minimum, development of preventive maintenance requirements and schedules.
- Track and assess failures of RISC-3 SSCs through the corrective action program that includes those actions outlined in Section 7.3 of this report.

9. REFERENCES

1. "10 CFR 50.69 SSC Categorization Guideline," NEI 00-04, Revision 0, Nuclear Energy Institute, February 2005.

2. "Guidelines For Categorizing Structures, Systems, And Components In Nuclear Power Plants According To Their Safety Significance" Regulatory Guide 1.201 For Trial Use, Nuclear Regulatory Commission, June 2004.

3. "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Regulatory Guide 1.147, Revision 14), Nuclear Regulatory Commission, August 2005.

4. "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," ASME Code Case N-660, July 2002.

5. "Wolf Creek Generating Station Probabilistic Risk Assessment," Revision 4, WCNOC, 1998.

6. "Wolf Creek Individual Plant Examination for External Events (IPEEE) Summary Report," TR-95-0015 W01, WCNOC, June 1995.

7. "WinNUPRA 2.1 PRA Model Calculator," Scientech Inc., 2001.

8. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 400-R-92-001, U. S. Environmental Protection Agency, 1991.

9. "Design Basis Program Guidelines," NEI-97-04, Revision 1, Nuclear Energy Institute, February 2001.

10. "Guidelines for Managing NRC Commitment Changes," NEI 99-04, Revision 0, Nuclear Energy Institute, July 1999.

Appendix A: Differences in Guidance Used for WCGS IDP and Endorsed Versions of Guidance

NEI 00-04

The WCGS IDP was conducted with the April 2004 version of the NEI 00-04 guidance, which was known as the "Final Draft." This version was transmitted to the NRC from NEI on April 14, 2004 (designated ML041120208 in the NRC's ADAMS document retrieval system). The latest version of NEI 00-04 is known as Revision 0 and was transmitted to the NRC from NEI on February 2, 2005. The primary differences between the version used in the WCGS IDP and the latest version are shown in Table A-1. Only those differenced that could impact the categorization process used at WCGS are shown in Table A-1.

Table A-1 shows both the "Final Draft" and the "Revision 0" guidance and a discussion the impact of the changed guidance on the WCGS categorization process.

Table A-1: Impact of Changes in NEI 00-04 on the WCGS Categorization

NEI 00-04 Section	New Guidance	Impact on WCGS Categorization
1.5	Clarifies the parts of the categorization process in which SSCs identified as HSS cannot be changed by the IDP versus those parts of the categorization process in which SSCs identified as potentially HSS can be considered for categorization as LSS by the IDP.	NO IMPACT - There were no cases during the WCGS IDP where a candidate HSS SSC was considered for categorization as LSS. WCNOC would follow the latest NEI guidance in future considerations.
1.5	Summary information added on the role of the various PRA assessments (internal events, seismic, fire, etc.) in the risk categorization process.	NO IMPACT – The additional information is consistent with the detailed information in later sections which remained unchanged from the earlier version.
Table 1-1	Summary information added on the types of PRA assessments (internal events, seismic, fire, etc.) that can be used in the risk categorization process.	NO IMPACT – The information in Table 1-1 is consistent with the detailed information in later sections which remained unchanged from the earlier version.
2.0	Information added related to the final 50.69 rule.	NO IMPACT – The information was added for completeness. The relevant parts of the final 50.69 rule related to the categorization process did not change the categorization process in NEI 00-04.
3 to 12	References to specific paragraphs of the 50.69 rule were added throughout Sections 3 through 12 to show how the NEI 00-04 guidance satisfies the rule requirements	NO IMPACT – The new information provides additional information but does not change the categorization process described in NEI 00-04.
3.3.1	References to Regulatory Guide 1.200 and other requirements for a quality PRA were added.	NO IMPACT – This report describes the categorization process used at WCGS and does not discuss the quality aspects of the PRA that would be used in an actual categorization exercise. The WCGS PRA will be updated to acceptable scope and quality levels and information will be provided to NRC to support the WCNOC LAR for implementing 50.69 at WCGS.
5.3	Added guidance on seismic PRA assessments for SSCs screened out of the seismic PRA because of their seismic robustness.	NO IMPACT – This applies to quantitative seismic PRAs. The seismic risk assessment for WCGS uses the SMA approach and not the quantitative PRA.
5.4	Added guidance regarding LSS for qualitative external events risk assessments for screened and unscreened scenarios.	NO IMPACT – The categorization for the two systems was reviewed and it was determined that no SSCs were credited in either screened or unscreened scenarios.
6.1	Clarification was added regarding the use of Figure 6-1 for Defense In Depth assessments.	NO IMPACT – The method described in the new guidance corresponds to the method used in the WCGS categorization for the two systems.

Table A-1: Impact of Changes in NEI 00-04 on the WCGS Categorization (cont.)

NEI 00-04 Section	New Guidance	Impact on WCGS Categorization
6.2	Am additional consideration was added for ISLOCA Defense In Depth to include the potential for an SSC to have an ISLOCA mitigation role.	NO IMPACT – The categorization for the two systems was reviewed and it was determined that no SSCs could be credited for mitigation of an ISLOCA event.
8.0	Additional clarification was added to explain the purpose of the risk sensitivity study and the basis for the method discussed in Section 8.1 versus the 50.69 rule requirements.	NO IMPACT – No RISC-3 SSCs from the two systems were modeled in the PRA. Therefore, a risk sensitivity assessment was not performed.
8.1	The considerations for maintaining the validity of the sensitivity study following initial categorization were deleted from Section 8 and added to Section 12	NO IMPACT – See discussion under Section 12 addition.
9.1	The makeup of the IDP was changed to delete "Licensing" and to specifically include "Safety Analysis" as a separate discipline.	NO IMPACT – The WCGS IDP included both Licensing and Safety Analysis disciplines. As noted in Section 2.3.7 of this document, WCNOC does not intend to include a Licensing representative on future IDPs.
9.2.2	The issues for the IDP consideration for candidate low safety significant SSCs have been changed based on experience during the pilot applications at WCGS and Surry. Additionally, some changes were made to assure consistency between the NEI 00-04 and the ASME Code Case guidance.	NO IMPACT – The IDP considerations for WCGS were reviewed and it was determined that none of the changes would impact the IDP decisions for either system. Also, none of the issues raised during the December 2003 IDP at WCGS at that were resolved with the April 2004 version of NEI 00-04 were changed in such as way as to re-open any of the issues.
9.2.2	The provision to use a plant condition monitoring program as a reason to override the other considerations was deleted by moving this consideration from the end of 9.2.2 to one of the actual considerations.	NO IMPACT – This provision was not used in the WCGS IPD considerations.
9.2.2	The provision to use historical data to show that failure modes are unlikely to occur and such failure modes can be detected in a timely fashion was deleted from the guidance.	NO IMPACT – This provision was not used in the WCGS IPD considerations.
11.1	Clarification was added to refer to 10 CFR 50.2, NRC Regulatory Guide 1.186, and NEI 97-04, Rev 1 for definitions of design basis commitments	NO IMPACT – The categorization process described in this report, which was performed under the April 2004 version of NEI 00-04, does not address maintenance of design commitments. Therefore the change in NEI guidance is has no impact. The WCNOC method of addressing this guidance item is discussed in the main section of this report.

Table A-1: Impact of Changes in NEI 00-04 on the WCGS Categorization (cont.)

NEI 00-04 Section	New Guidance	Impact on WCGS Categorization
12.1	Clarification was added related top scheduling of periodic reviews.	NO IMPACT – The categorization process described in this report, which was performed under the April 2004 version of NEI 00-04, does not address periodic reviews. Therefore the change in NEI guidance is has no impact. The WCNOC method of addressing this guidance item is discussed in the main section of this report.
12.2	Clarification was added related to phased implementation and the impact of later categorization results on earlier categorizations.	NO IMPACT – The categorization process described in this report, which was performed under the April 2004 version of NEI 00-04, was the initial categorization. Therefore the change in NEI guidance has no impact. The WCNOC method of addressing this guidance item is discussed in the main section of this report.
12.3	Clarification was added related to monitoring failures versus Maintenance Rule monitoring for categorized SSCs.	NO IMPACT – The categorization process described in this report, which was performed under the April 2004 version of NEI 00-04, does not address monitoring. Therefore the change in NEI guidance has no impact. The WCNOC method of addressing this guidance item is discussed in the main section of this report.
12.4	Clarification was added related to performance monitoring and corrective action for failures of categorized SSCs.	NO IMPACT – The categorization process described in this report, which was performed under the April 2004 version of NEI 00-04, does not address monitoring or corrective actions. Therefore the change in NEI guidance has no impact. The WCNOC method of addressing this guidance item is discussed in the main section of this report.

ASME Code Case N-660

ASME Code Case N-660 Revision 0 was approved by ASME in 2002 and was later endorsed by the NRC in Regulatory Guide 1.147, Revision 14. The first WCGS IDP was conducted in December of 2003 and used this version of the Code Case. Several issues with the Code Case were identified at that IDP and a revision was proposed to ASME. An April 2004 draft of a revision to the ASME Code Case N-660, was used in the subsequent WCGS IDP and was the basis for the categorization process described in Revision 0 of this report.

As a result of the NRC review of the passive categorization process proposed for use by WCNOC, additional changes were made to Code Case. The differences between the NRC approved categorization process and the ASME Code Case N-660 as approved by ASME in 2002 are shown in Table A-2. In addition, a markup of the approved ASME Code Case N-660, to show these changes in the context of the Code Case language, is provided following Table A-2.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
-1320	"Personnel with expertise in the following disciplines shall be included in the classification process. *(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."	Replaced with "(a) An Integrated Decisionmaking Panel (IDP) shall use the information and insights compiled in the initial categorization process and combine that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of functions/SSCs. (b) The designated as members of the IDP shall have joint expertise in the following fields: - Plant Operations (SRO qualified), - Design Engineering, - Safety analysis, - Systems Engineering, and - Probabilistic Risk Assessment. (c) Requirements for ensuing adequate expertise levels and training of IDP members in the categorization process shall be established. (d) To the extent possible, the classification of pressure retaining and support items in a system should be performed by the same IDP members as the categorization of active SSCs in that system."	Clarification of the process used for the categorization of pressure retaining and support items. An initial categorization of pressure retaining and support items can be performed by an engineering function. The IDP, composed of the members with expertise in the disciplines identified in the original paragraph -1320, then considered the initial categorization, along with other information from their respective disciplines, to finalize the categorization. This method results in a categorization process for classifying pressure retaining and support items that is similar to that used for active SSCs. This helps to ensure consistent consideration of information used the two categorization processes.
-9000	Definition of high-safety-significant function	Added to end of definition – "or from other relevant information (e.g., defense in depth considerations)"	Added to consider defense in depth in determining the safety significance of a function.
-9000	N/A	Added new term and definition, "Plant features – systems, structures, and components that can be used to prevent or mitigate an accident"	Plant features terminology added to Code Case relative to operator and possible automatic actions
-9000	Definition of PRA, "a qualitative and quantitative assessment…"	Changed to read, "an assessment…"	Changed to be consistent with the ASME PRA Standard.
-9000	Definition of spatial effects, "A failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement or flooding."	Changed to read, "A failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, loss of inventory due to draining of a tank, or flooding."	Including other possible forms of spatial effects.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-1.0	N/A	Added figure[1] illustrating the modified RISC methodology process, including scope identification, consequence evaluation, consequence categorization, classification considerations, and final classification definitions.	Figure added to provide high level overview of RISC methodology process. New process calls for all segments to be included in the consequence evaluation to determine high, medium, low or none consequence category. Then only the non-high category segments would be considered in the classification considerations of I-3.2.2(b) – previously I-3.1.3.
	"Once categorized, the safety significance of each piping segment is identified"	Change to read, "Once categorized, the safety significance of each piping segment is identified Figure I-1 illustrates the RISC methodology presented in the following sections.	Text refers to the new figure.
I-2.0	"The owner shall define the boundaries included in the scope of the RISC evaluation process."	Changed to read, "The owner shall define the boundaries included in the scope of the RISC evaluation process subject to the constraints in paragraph 50.69(c)(1)(v) that the categorization must be performed for entire systems. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply."	The second and third sentences added for clarification of the scope of items to be evaluated, consistent with -1200
I-3.0, Title	"Consequence Assessment"	Changed to read, "Evaluation of Risk Informed Safety Classifications"	For clarification to meet Figure I-1.
I-3.0, 1st Paragraph	"Piping segments can be grouped based on common conditional consequence…"	Changed to read, "All pressure retaining items, including supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common conditional consequence…"	For clarification of the scope of components to be evaluated.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.0, 1st Paragraph	"Additionally, information shall be collected for each piping segment that is not modeled in the PRA, but considered relevant to the classification (e.g., information regarding design basis accidents, shutdown risk, containment isolation, flooding, fires, seismic conditions)."	Changed to read, "Additionally, information considered relevant to the classification shall be collected for each piping segment (e.g., information regarding design basis accidents, at-power risk, shutdown risk, containment isolation, flooding, fires, seismic conditions, etc.). Consistent with 50.69(c)(1)(ii), the classification must address initiating events and plant operating modes."	Clarifies requirement to collect relevant information for ALL piping segments, not just those modeled in the PRA.
I-3.1.1, 1st Sentence	"Potential failure modes for each piping segment shall be identified…"	Changed to read, "Potential failure modes for each system or piping segment shall be identified…"	Clarify that evaluation should consider system level failure modes as well as piping segment failure modes.
I-3.1.1(c), Indirect Effects	"These include spatial interactions such as pipe whip, jet spray, and loss of inventory effects (e.g., draining of a tank)."	Changed to read, "A failure consequence affecting other systems or components, such as spatial effects."	To be consistent with glossary term for spatial effect.
I-3.1.1(d), Initiating Events	"These are identified using a list of initiating events from any existing plant specific Probabilistic Risk Assessment (PRA) or Individual Plant Examination (IPE) and the Owner's Requirements."	Changed to read, "For systems or piping segments that are modeled either explicitly or implicitly in any existing plant-specific Probabilistic Risk Assessment (PRA), any applicable initiating event is identified using a list of initiating events from that PRA."	Clarify source of initiating events.
I-3.1.2, 3rd sentence	"… (high, medium, low)..."	Changed to read, "… (high, medium, low, or none)…"	"None" is one of the four consequence categories which can be assigned in I-3.1.
I-3.1.2	N/A	Added text, "In assessing the appropriate consequence category, risk information for all initiating events, including fire and seismic, should be considered. To capture the risk importance from initiating events for which no quantitative PRA is available, any piping segment supporting a safe shutdown pathway would be classified as HSS."	This statement was added to help clarify Section I-3.0 when considering other relevant information.
I-3.1.2(a)(1)	"The initiating event shall be placed in one of the categories in Table I-1."	Changed to read, "The initiating event shall be placed in one of the Design Basis Event Categories in Table I-1."	More clearly defined what "category" means relative to Table I-1.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.1.2(a)(1)	"... updated final safety analysis report, PRA, or IPE shall be included"	Changed to read, "... updated final safety analysis report or PRA shall be included"	Removed IPE because it was felt that the IPE is no longer relevant for this application and does not provide any additional information in this area.
I-3.1.2(b)(1)	"Frequency of challenge that determines how often the mitigating function of the system is called upon. This corresponds to the frequency of initiating events that require the system operation."	Changed to read, "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."	Clarified to include functions other than simply mitigating functions and all events as opposed to only initiating events.
I-3.1.2(b)(3)	"Exposure time shall be obtained from Technical Specification limits."	Sentence deleted	Deletion made because it was redundant to the 2nd sentence.
I-3.1.2(b)(3)	"In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5."	Moved out from (b)(3) to directly under (b)	Clarification; this statement applies to all of (b) and not only (3) for Exposure Time.
I-3.1.2(d)	"The above evaluations determine failure importance relative to core damage."	Changed to read, "The above evaluations determine failure importance relative to core damage or the plant's capability to reach or maintain safe shutdown conditions."	Added consistent with the changes made to I-3.1.2(b).
I-3.1.3, 3.1.4, & 3.1.5	These three sections have been removed and categorization guidance has been moved to the revised Sections I-3.2.2(b) and (c). In most cases the old guidance in I-3.1.3 through 3.1.5 and the new guidance in I-3.2.2(b) and (c) is identical or very similar. The disposition of each paragraph for I-3.1.3 through I-3.1.5 is provided directly below. The original intent of section was to provide additional considerations for segments not modeled in the PRA. However, the grouping of components into piping segments and the use of surrogate components in the PRA provides quantitative evaluations for each piping segment. The intent of this section now is to provide further considerations for piping segments with MEDIUM, LOW, or NONE consequence categories. The new process calls for all segments to be created and assigned a consequence category according to the guidance in Sections I-3.1.1 & I-3.1.2. For those segments with a consequence category of MEDIUM, LOW, or NONE, the user must then use the guidance in I-3.2.2(b), which is based on the old considerations in Sections I-3.1.3, 3.1.4, and 3.1.5, to assign a final high or low safety significance. See the table entries for specific changes made to Sections I-3.1.3, 3.1.4, and 3.1.5.		
Old I-3.1.3; New I-3.2.2(b)	All	Questions changed such that all TRUE responses will support LSS and any single FALSE response will support HSS.	For consistency with NEI 00-04 process where a TRUE response to similar questions supports a LSS finding.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.1.3(a)(1)	"Failure of the piping segment will significantly increase the frequency of an initiating event, including those initiating events originally screened out in the PRA, such that the CDF or large early release frequency (LERF) would be estimated to increase by more than 10-6/yr or 10-7/yr, respectively."	Consideration deleted	Redundant to the considerations in I-3.1.1 and I-3.1.2 when determining failure consequences and consequence category.
I-3.1.3(a)(2)	"Failure of the piping segment will compromise the integrity of the reactor coolant pressure boundary as defined in –1200(b)."	Consideration deleted	All reactor coolant pressure boundary segments are ranked high safety significant per -1200(b).
I-3.1.3(a)(3)	"Even when considering operator actions used to mitigate an accident, failure of the piping segment will fail a high safety significant function."	Consideration changed and moved to new Section I-3.2.2(b)(1), "Even when taking credit for plant features and operator actions, failure of the piping segment will not fail a high safety-significant function."	Added ability to credit plant features and operator actions when evaluating failure impact on high safety significant functions. Footnote provided for credible operator actions (see below).
I-3.1.3(a)(4)	"Failure of the piping segment will result in failure of other safety-significant piping segments, e.g., through indirect effects."	Consideration changed and moved to new Section I-3.2.2(b)(2), "Failure of the piping segment will not result in failure of a high safety-significant piping segment, e.g., through indirect effects."	Consistent with definition of HSS function
I-3.1.3(a)(5)	"Failure of the piping segment will prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions."	Consideration changed and moved to new Section I-3.2.2(b)(3), "Failure of the piping segment will not prevent the plant reaching or maintaining safe shutdown conditions."	Added ability to credit plant features and operator actions when evaluating failure impact on shutdown conditions. Footnote provided for credible operator actions (see below).
I-3.1.3(b)	"In addition to being HSS in terms of their contribution to CDF or LERF, piping segments might also be HSS in terms of other risk metrics or conditions. Therefore, the following conditions shall be evaluated."	Consideration deleted	The new Section I-3.2.2(b) creates a single list of the considerations from I-3.1.3(a) and I-3.1.3(b). Therefore this lead-in to the considerations in I-3.1.3(b) is unnecessary.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.1.3(b)(1)	"The piping segment is a part of a system that acts as a barrier to fission product release during severe accidents."	Consideration deleted	This statement was too conservative to force all segments to be ranked as HSS given that just one segment in the entire system meets this criterion. The intent of this consideration is expressed in new subsections I-3.2.2(b)(6) and (11).
I-3.1.3(b)(2)	"The piping segment supports a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines."	Consideration changed and moved to new Section I-3.2.2(b)(4), "The piping segment does not individually support a sole means for successful performance of operator actions addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines required to mitigate an accident or transient, including instrumentation and other equipment associated with the required actions."	The original statement was too limiting for any segment supporting functions addressed in the EOPs or SAMGs. The term 'significant' was too vague. New statement clarifies the interpretation and allows for reasonable consideration of plant features and operator actions. However, the new language assures that the redundant or alternate means are available in the EOP or SAMG to address the function.
I-3.1.3(b)(3)	"Failure of the piping segment will result in unintentional releases of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100."	Consideration changed and moved to new Section I-3.2.2(b)(6), "Failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions."	The off-site emergency response and protective actions limits are more limiting compared to those in 10 CFR Part 100.
I-3.1.4	"Maintain Defense in Depth. When categorizing piping segments LSS, the RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant Specific Changes to the Licensing Basis." Dated July 1998."	Entire section I-3.1.4 deleted. See new Section I-3.2.2(b)(7-11) for replacement text.	Replacement text in I-3.2.2(b)(7-11) is not intended to change the content – changes were made to be consistent with NEI 00-04 defense in depth considerations.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.1.5	"Maintenance of Adequate Safety Margins. When categorizing piping segments LSS, the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety margin shall be incorporated when determining performance characteristics and parameters, e.g., piping segment, system, and plant capability or cusses 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 safety 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."	Entire section I-3.1.5 deleted. See new Section I-3.2.2(c) for replacement text.	Replacement text in I-3.2.2(c) is not intended to change the content.
I-3.2	N/A	Added as first sentence, "Risk Informed Safety Classification is determined by considering the Consequence Category."	Added to clarify intent of I-3.2.
I-3.2.2(b)	Rather than referring to Sections I-3.1.3, I-3.1.4, and I-3.1.5, new considerations have been provided as listed above. The process requires the user to evaluate the additional considerations for any segment with consequence category Medium, Low, or None. To improve the process, the additional considerations were moved into this section from I-3.1.3, I-3.1.4, and I-3.1.5.		

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.2.2(b)	"Piping segments determined to be Medium consequence category in any table by the consequence evaluation (I-3.1.1) and (I-3.1.2) shall be determined HSS or LSS by considering the RISC evaluation and the other relevant information (I-3.1.3, I-3.1.4, and I-3.1.5) provided for determining classification."	Changed text to read, "Piping segments determined to be Medium, Low or None (no change to base case) consequence category in any table by the consequence evaluation in Section I-3.1 shall be determined HSS or LSS by considering the other relevant information for determining classification. The following conditions shall be evaluated and answered true or no true. If any of the following eleven (11) conditions are not true, HSS should be assigned."	Changed to include Low and None consequence category segments for consideration and removed reference to deleted Sections.
I-3.2.2(b)	"Any piping segment initially determined to be a Medium consequence category and that is subject to a known active degradation mechanism shall be classified HSS."	Consideration deleted	This is too restrictive. Consideration should be given to existing plant programs that may affect the ability to prevent a pipe segment from failing given a known active degradation mechanism
I-3.2.2(b)	N/A	Added the following sentence just before I-3.2.2(1); "The following conditions shall be evaluated and answered true or not true:"	Clarification provided to answering the additional considerations as true or not true. If any one of the eleven considerations is not true then the segment shall be assigned HSS, otherwise it can be assigned LSS.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.2.2(b), footnote	N/A	Added footnote to "operator actions" as follows; "To credit operator actions, the following criteria must be met: • There must be an alarm or clear indication of the failure. • A procedure must direct the response to the alarm or indication. • Equipment activated to alleviate the condition must not be affected by the failure. • There must be sufficient time to perform the compensatory action."	Words paraphrased from Supplement 2, Rev 1 of WCAP-14572, Rev 1, the Pressurized Water Reactor Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications. The guidance is provided for expert panel members when relying on operator actions to make decisions regarding safety significance.
I-3.2.2(b)(1)	N/A	Added new Section I-3.2.2(b)(1), "Even when taking credit for plant features and operator actions, failure of the piping segment will not fail a high safety-significant function."	Based on original Section I-3.1.3(a)(3) – see table entry above.
I-3.2.2(b)(2)	N/A	Added new Section I-3.2.2(b)(2), "Failure of the piping segment will not result in failure of a high safety-significant piping segment, e.g., through indirect effects."	Based on original Section I-3.1.3(a)(4) – see table entry above.
I-3.2.2(b)(3)	N/A	Added new Section I-3.2.2(b)(3), "Failure of the piping segment will not prevent the plant reaching or maintaining safe shutdown conditions."	Based on original Section I-3.1.3(a)(5) – see table entry above.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.2.2(b)(4)	N/A	Added new Section I-3.2.2(b)(4), "The piping segment does not individually support a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines, with no redundancy or alternate means of support."	Based on original Section I-3.1.3(b)(2) – see table entry above.
I-3.2.2(b)(5)	N/A	Added new Section I-3.2.2(b)(5), "The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program)."	In response to removal of statement regarding treatment of Medium consequence category segments subject to a known active degradation mechanism (see above I-3.2.2(b)).
I-3.2.2(b)(6)	N/A	Added new Section I-3.2.2(b)(6), "Failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions."	Based on original Section I-3.1.3(b)(3) – see table entry above.
I-3.2.2(b) between (6) and (7)	N/A	Added, "The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated November 2002. Defense-in-depth is maintained if:"	Based on original Section I-3.1.4 – see table entry above.
I-3.2.2(b)(7)	N/A	Added, "A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation."	Taken from Reg Guide 1.174.
I-3.2.2(b)(8)	N/A	Added, "Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided."	Taken from Reg Guide 1.174.

N-660, R0 Section	ASME Code Case N-660 Revision 0	Proposed Changes to ASME Code Case N-660 Revision 0 For PWROG Passive Categorization	Basis for Change
I-3.2.2(b)(9)	N/A	Added, "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)."	Taken from Reg Guide 1.174.
I-3.2.2(b)(10)	N/A	Added, "Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed."	Taken from Reg Guide 1.174.
I-3.2.2(b)(11)	N/A	Added, "Independence of fission-product barriers is not degraded."	Taken from Reg Guide 1.174.
I-3.2.2	N/A	Added sentence following I-3.2.2(11); "If any of the above eleven (11) conditions are not true, HSS should be assigned.	Statement added to instruct expert panel which ranking to assign based on the answers to the eleven considerations. Also consistent with NEI 00-04.
I-3.2.2(c)	N/A	Changed first sentence from original Section I-3.1.5 to read, "If LSS has been assigned from I-3.2.2(b), then the RSC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data." Added 2nd, 3rd, and 4th sentences from the original Section I-3.1.5 without change. Added new sentence, "If LSS has been assigned from I-3.2.2(b) and at least one of the above safety margin conditions are true, then LSS should be assigned; if both of the above safety margin conditions are not true, then HSS shall be assigned."	Original Section I-3.1.5 restated for clarity – no intended change in methodology. Moved to Section I-3.2.2 for consistency.
Table I-1	Table entry for Design Basis Event Category I and Consequence Category was "N/A"	"N/A" changed to "None"	None is a recognized Consequence Category that must then be processed through the additional considerations in I-3.2.2. N/A indicated that there was no consequence category and the pipe segment could be categorized as LSS without additional considerations.

Note 1 – Figure I-1, Risk-Informed Safety Classification Process

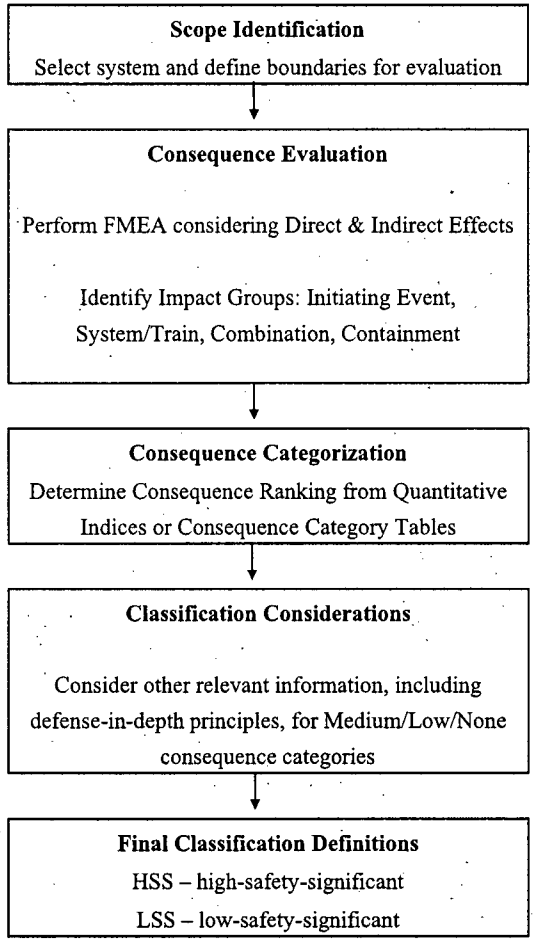

Figure I-1: Risk-Informed Safety Classification Process

Passive Categorization Guidance for PWROG Topical Report WCAP-16308-NP

-1000 SCOPE AND RESPONSIBILITY

-1100 Scope

This Case provides a process for determining the Risk-Informed Safety Classification (RISC) for use in risk-informed repair/replacement activities. The RISC process of this Case may be applied to any of Class 1, 2, 3, or non-class[1] pressure-retaining items or their associated supports, except core supports, in accordance with the risk-informed safety classification criteria established by the regulatory authority having jurisdiction at the plant site.

-1200 Classifications

(a) The RISC process is described in Appendix I of this Case. Pressure retaining and component support items shall be classified High Safety Significant (HSS) or Low Safety Significant (LSS). However, because this classification is to be used only for repair/replacement activities, failure potential is conservatively assumed to be 1.0 in determining a consequence category in Appendix I. These classifications might not be directly related to other risk-informed applications.

(b) Class 1 items that are part of the reactor coolant pressure boundary except as provided in paragraphs (c)(2)(i) and (c)(2)(ii) of Title 10 of the U.S. Code of Federal Regulations (10 CFR), Part 50.55a shall be classified High Safety Significant (HSS). For items that are connected to the reactor coolant pressure boundary, as defined in paragraph 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), the RISC process of (a) should be applied.

-1300 OWNER'S RESPONSIBILITY

-1310 Determination of Classification

The responsibilities of the Owner shall include determination of the appropriate classification for the items identified for each risk-informed repair/replacement activity, in accordance with Appendix I of this Case. The Owner shall ensure that core damage frequency (CDF) and large early release frequency (LERF) are included as risk metrics in the RISC process.

-1320 Required Disciplines

(a) An Integrated Decisionmaking Panel (IDP) shall use the information and insights compiled in the initial categorization process and combine that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of functions/SSCs.

(b) The designated as members of the IDP shall have joint expertise in the following fields:

- Plant Operations (SRO qualified),
- Design Engineering,
- Safety analysis,
- Systems Engineering, and
- Probabilistic Risk Assessment.

(c) Requirements for ensuing adequate expertise levels and training of IDP members in the categorization process shall be established.

[1] Non-class items are items not classified in accordance with IWA-1320.

(d) To the extent possible, the classification of pressure retaining and support items in a system should be performed by the same IDP members as the categorization of active SSCs in that system.

-1330 Adequacy of the PRA

The Owner is responsible for demonstrating adequacy of any PRA used as the basis for this process. All deficiencies identified shall be reconciled during the analysis to support the RISC process. The resolution of all PRA issues shall be documented.

-9000 GLOSSARY

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)

core damage – uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated and involving enough of the core to cause a significant release

failure – an event involving leakage, rupture, or a condition that would disable the ability of an item to perform its intended safety function

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

high-safety-significant function – a function that has been determined to be safety significant from traditional plant risk-assessment evaluations of core damage or large early release events (e.g., evaluations performed to support the Maintenance Rule - 10 CFR 50.65 or from other relevant information (e.g., defense in depth considerations)

initiating event (IE) – any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event, such as a transient or LOCA within the plant. 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

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

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

plant features – systems, structures, and components that can be used to prevent or mitigate an accident

probabilistic risk assessment (PRA) – an 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 (also referred to as a probabilistic safety assessment, PSA)

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

risk metrics – a determination of what activity or conditions produce the risk, and what individual, group, or property is affected by the risk

spatial effect – a failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, loss of inventory due to draining of a tank, or flooding

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 (mission time), to ensure that the safety functions are satisfied

APPENDIX I RISK-INFORMED SAFETY CLASSIFICATION (RISC) PROCESS

I-1.0 INTRODUCTION

This Appendix provides the risk-informed process used to determine Risk-Informed Safety Classification (RISC) for use in risk-informed repair/replacement activities. This RISC process is based on conditional consequence of failure. The process provides a conservative assessment of the importance of an item. This process divides each selected system into piping segments that are determined to have similar consequence of failure. These piping segments are categorized based on the conditional consequence. Once categorized, the safety significance of each piping segment is identified. Figure I-1 illustrates the RISC methodology presented in the following sections.

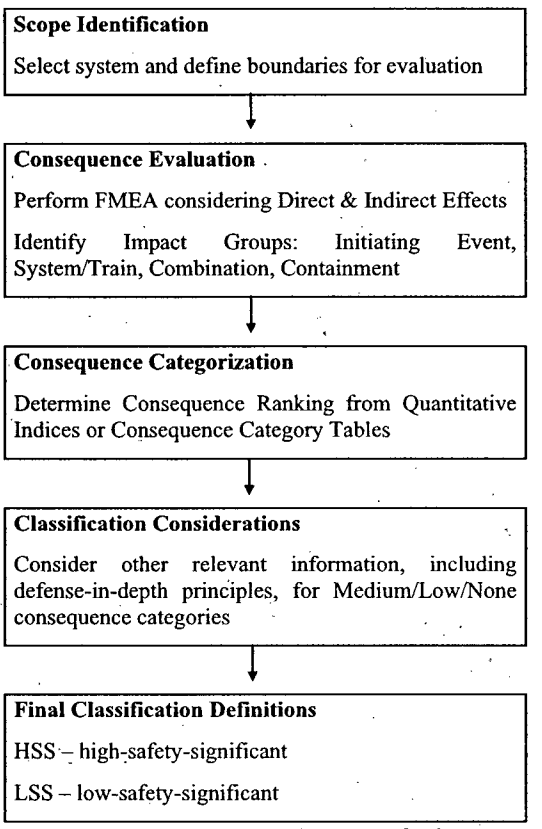

Figure I-1: Risk-Informed Safety Classification Process

I-2.0 SCOPE IDENTIFICATION

The Owner shall define the boundaries included in the scope of the RISC evaluation process subject to the constraints of 50.69(c)(1)(v) that the categorization must be performed for entire systems. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply.

I-3.0 EVALUATION OF RISK-INFORMED SAFETY CLASSIFICATIONS

All pressure retaining items, including supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common conditional consequence (i.e., given failure of the piping segment). To accomplish this grouping, the direct effects, and indirect effects shall be assessed for each piping segment. Additionally, information considered relevant to the classification shall be collected for each piping segment (e.g., information regarding design basis accidents, at-power risk, shutdown risk, containment isolation, flooding, fires, seismic conditions, etc.). Consistent with 50.69(c)(1)(ii), the classification must address all initiating events and plant operating modes.

I-3.1 Consequence Evaluation

I-3.1.1 Failure Modes and Effects Analysis (FMEA). Potential failure modes for each system or piping segment shall be identified, and their effects shall be evaluated. This evaluation shall consider the following:

(a) Pressure Boundary Failure Size. The consequence analysis shall be performed assuming a large pressure boundary failure for piping segments. Alternatively, the consequence analysis can be performed assuming a smaller leak, when

(1) a smaller leak is more conservative; or

(2) when a small leak can be justified through a leak-before-break analysis in accordance with the criteria specified in NUREG-1061, Volume 3; 10CFR50, Appendix A, General Design Criterion 4; or

(3) it can be documented that plant configuration precludes the possibility of a large pressure boundary failure.

(b) 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 or by operator action.

(c) Indirect Effects. A failure consequence affecting other systems or components, such as spatial effects.

(d) Initiating Events. For systems or piping segments that are modeled either explicitly or implicitly in any existing plant-specific Probabilistic Risk Assessment (PRA), any applicable initiating event is identified using a list of initiating events from that PRA.

(e) System Impact or Recovery. The means of detecting a failure, and the Technical Specifications associated with the system and other affected systems. Possible automatic and operator actions to prevent a loss of system function.

(f) System Redundancy. The existence of redundancy for accident mitigation purposes.

I-3.1.2 Impact Group Assessment. The results of the FMEA evaluation for each piping system, or portion thereof, shall be classified into one of three impact groups: initiating event, system, or combination. Each piping system, or portion thereof, shall be partitioned into postulated piping failures that cause an initiating event, disable a system without causing an initiating event, or cause an initiating event and disable a system. The consequence category assignment (high, medium, low, or none) for each piping segment within each impact group shall be selected in accordance with (a) through (d) below. In assessing the appropriate consequence category, risk information for all initiating events, including fire and seismic, should be considered. To capture risk importance from initiating events for which no quantitative PRA is available, any piping segment supporting a safe shutdown pathway would be classified as HSS.

(a) 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:

(1) The initiating event shall be placed in one of the Design Basis Event Categories in Table I-1. All applicable design basis events previously analyzed in the Owner's updated final safety analysis report or PRA shall be included.

(2) Breaks that cause an initiating event classified as Category I (routine operation) need not be considered in this analysis.

(3) For piping segment 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 I-5. The quantitative index for the initiating event impact group (CCDP) is the ratio of the core damage frequency due to the initiating event to the initiating event frequency.

(b) System Impact Group Assessment. The consequence category of a failure that does not cause an initiating event, but degrades or fails a system essential to prevention of core damage shall be based on the following:

(1) 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.

(2) 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 systems.

(3) 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 Allowed Outage Time, as defined in the plant Technical Specification. Consequence categories shall be assigned in accordance with Table I-2 as High, Medium, or Low. Frequency of challenge is grouped into design basis event categories II, III, and IV. The Owner or his designee shall ensure that the quantitative basis of Table I-2 (e.g., one full train unavailability approximately 10^{-2}) is consistent with the failure scenario being evaluated.

In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5.

(c) 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 I-3. The Owner or his designee shall ensure that the quantitative basis of Table I-3 (e.g., one full train unavailability approximately 10^{-2}) is consistent with the pipe failure scenario being evaluated. The consequence category is a function of two factors:

(1) Use of the system to mitigate the induced initiating event;

(2) Number of unaffected backup systems or trains available to perform the same function.

In lieu of Table I-3, quantitative indices may be used to assign consequence categories in accordance with Table I-5.

(d) Containment Performance. The above evaluations determine failure importance relative to core damage or the plant's capability to reach or maintain safe shutdown conditions. Failure shall also be evaluated for its effect on containment performance. This shall be accomplished by addressing two issues, both of which are based on an approximate conditional probability value of not greater than 0.1 between the CCDP and the likelihood of large early release from containment. If there is no margin, i.e., conditional probability of a large early release due to core damage is greater than 0.1, the assigned consequence category shall be increased one level. The two issues are described as follows:

(1) CCDP values for initiating events and safety functions are evaluated to determine if the potential for large early release due to containment failure requires the consequence category to be increased.

(2) The effect on containment isolation is evaluated. If there is a containment barrier available, the consequence category from the core damage assessment is retained. If there is no containment barrier or the barrier failed in determining the consequence category from the core damage assessment, some margin in the core damage consequence category assignment must be present for it to be retained. For example, if the CCDP for core damage is less than 10^{-5}, i.e., a Medium consequence assignment, and there is no containment barrier, the Medium consequence assignment is retained, because there is 0.1 margin to the High consequence category threshold, i.e., 10^{-4}. However, if the CCDP for core damage is $5x10^{-5}$, i.e., a Medium consequence assignment, and there is no containment barrier, the consequence category is increased to High, because the margin to the High consequence category threshold, i.e., 10^{-4}, is less than 0.1. Table I-4 shall be used to assign consequence categories for those piping failures that can lead to a LOCA outside containment. In lieu of using Table I-4, quantitative indices may be used to assign consequence categories in accordance with Table I-5 with each range lowered one order of magnitude, e.g., not less than 10^{-5} is High.

I-3.2 Classification

Risk Informed Safety Classification is determined by considering the Consequence Category.

I-3.2.1 Final Risk-Informed Safety Classification. Piping segments may be grouped together within a system, if the consequence evaluation (I-3.1) determines the effect of the postulated failures to be the same. The Risk-Informed Safety Classification shall be as follows:

Classification Definitions

HSS – Piping segment considered high-safety-significant

LSS – Piping segment considered low-safety-significant

I-3.2.2 Classification Considerations.

(a) Piping segments determined to be a High consequence category in any table by the consequence evaluation (I-3.1.1 and I-3.1.2) shall be considered HSS. The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.

(b) 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 I-3.1 shall be determined HSS or LSS by considering the other relevant information for determining classification. The following conditions shall be evaluated and answered true or not true. If any of the following eleven (11) conditions are not true, HSS should be assigned.

(1) Even when taking credit for plant features and operator actions[2], failure of the piping segment will not fail a high-safety-significant function.

[2] To credit operator actions, the following criteria must be met:
- There must be an alarm or clear indication of the failure.
- A procedure must direct the response to the alarm or indication.
- Equipment activated to alleviate the condition must not be affected by the failure.
- There must be sufficient time to perform the compensatory action

(2) Failure of the piping segment will not result in failure of a high safety-significant piping segment, e.g., through indirect effects.

(3) Failure of the piping segment will not prevent the plant reaching or maintaining safe shutdown conditions.

(4) The piping segment does not individually support a sole means for successful performance of operator actions addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines required to mitigate an accident or transient, including instrumentation and other equipment associated with the required actions.

(5) The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program).

(6) Failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions.

The RISC process shall demonstrate that the defense-in-depth philosophy is maintained. Defense-in-depth may be demonstrated by following the guidelines of U.S.N.R.C. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated November 2002. Defense-in-depth is maintained if:

(7) A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

(8) Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

(9) 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).

(10) Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.

(11) Independence of fission-product barriers is not degraded.

(c) If LSS has been assigned from I-3.2.2(b), then the RISC process shall verify that there are sufficient safety margins to account for uncertainty in the engineering analysis and in the supporting data. Safety 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 safety margins are maintained by:

(1) Ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or

(2) Ensuring that proposed revisions account for analysis and data uncertainty.

If LSS has been assigned from I-3.2.2(b) and at least one of the above safety margin conditions are true, then LSS should be assigned; if both of the above safety margin conditions are not true, then HSS shall be assigned.

(d) A component support or snubber shall have the same classification as the highest-ranked piping segment within the piping analytical model in which the support is included. The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.

I-4.0 Reevaluation of Risk-Informed Safety Classifications

New information may become available that alters the RISC for a piping segment. Such information may result from changes to the PRA, plant operation, or design of items. The Owner shall identify and verify the effect of the new information on the RISC assigned to the piping segment.

When it is determined that the new information affects the RISC, the Owner shall reevaluate the classification, using the same approach originally used to establish the RISC.

TABLE I-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	$\geq 10^{-1}$	Reactor Trip, Turbine Trip, Partial Loss of Feedwater	Low/ Medium
III	Infrequent Event	10^{-1} to 10^{-2}	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 I-3.1.2(a)(3)

TABLE I-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 AOT (≤ 1 week)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Short AOT (≤ 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 AOT (≤ 1 week)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short AOT (≤ 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 AOT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Short AOT (≤ 1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

Note: 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 I-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]
Initiating Event and More Than 2 Unaffected Trains of Mitigating Systems Available	Low[1]
Initiating Event and No Mitigating System Affected	N/A

Note 1: The higher classification of this table or Table I-1 shall be used.

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

9.1 PROTECTION AGAINST 9.2 LOCA OUTSIDE CONTAINMENT	9.3 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

Note 1: An example of Active Protection is a valve that needs to close on demand.

Note 2: An example of Passive Protection is a valve that needs to remain closed.

TABLE I-5: QUANTITATIVE INDICES FOR CONSEQUENCE CATEGORIES

CCDP or Quantitative Index, no units	Consequence Category
$\geq 10^{-4}$	High
$10^{-6} \leq$ value $< 10^{-4}$	Medium
$<10^{-6}$	Low
No change to base case	None

Appendix B: NRC's Request for Additional Information and PWROG Responses

Note that OG-07-459 Enclosure 2 was later revised as a result of the NRC's Safety Evaluation dated March 26, 2009. The final Table A-2 is included in Appendix A to this report.

PWROG
PWR Owners Group

Program Management Office
4350 Northern Pike
Monroeville, Pennsylvania 15146

October 18, 2007

OG-07-459

Mr. Biff Bradley
Nuclear Energy Institute
1776 I Street, NW
Washington, D.C. 20006-3708

Subject: PWR Owners Group
Transmittal of PWROG Responses to NRC Request For Additional Information on WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program - Categorization Process - Wolf Creek Generating Station (PA-SEE-0027)"

References: PWROG Letter, M. Dingler to B. Bradley (NEI), "Transmittal of Additional Information for WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program - Categorization Process - Wolf Creek Generating Station (PA-SEE-0027)", WOG-07-202, May 8, 2007.

Dear Biff,

Enclosed are the PWROG responses to the NRC's August 28th RAIs from their review of WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program - Categorization Process - Wolf Creek Generating Station." The PWROG response to the RAIs also includes WCAP revisions, as mark-ups. In addition, an updated Table A-2 for WCAP-16308-NP is provided based on changes from the version transmitted via Reference.

This information is consistent with the discussions with the NRC at the September 28, 2007 meeting to discuss the draft PWROG RAI responses.

NEI submitted WCAP-16308-NP to the NRC at the request of the PWROG on September 26, 2006. NEI may provide these RAI responses directly to the NRC.

Sincerely,

Frederick P. "Ted" Schiffley, II
Chairman, PWR Owners Group
PWR Owners Group

Enclosures

Mr. Biff Bradley, NEI
OG-07-459

October 18, 2007

cc: PWROG PMO
R. J. Lutz, Jr.
G.G. Ament

Enclosure 1

INTRODUCTION

At the September 28, 2007 public meeting (Accession No. ML071930260) to discuss the preliminary PWROG responses to the NRC's Requests for Additional Information (RAIs), the PWROG took an action to provide finalized RAI responses (including an additional RAI that was received after the meeting), provide an updated Table A-2 to WCAP-16308-NP, and to clarify the expectation for the scope of the NRC's review and safety evaluation of the PWROG Topical Report (TR) WCAP-16308-NP.

The RAI responses are provided in this document. The revised Table 2 to WCAP-16308-NP is provided in a separate document attached to the cover letter transmitting this document.

The PWROG expectation for the NRC review of WCAP-16308-NP is to obtain a clear safety evaluation which would allow licensees to take credit for the process used to categorize passive components in accordance with 10 CFR 50.69. This is consistent with the meeting summary from the February 6, 2007 public meeting (Ascension No. ML070440490). The PWROG believes that this includes the passive categorization process described in Section 4 of the TR and includes the IDP process used to finalize both the passive and the active categorization described in Section 3.7 or the TR. In addition, based on RAIs 10 and 11 below, the PWROG believes that the NRC's safety evaluation should also address the process used to satisfy 50.69 (d)(2) and (e) as discussed in the Section 7 and 8 of the TR as revised by the responses to RAIs 10 and 11. The active categorization described in Section 3 of the TR follows NEI 00-04 which has been endorsed by NRC in Regulatory Guide 1.200 and there is no need for further NRC review of this section, except Section 3.7 as discussed above.

NRC RAIs on WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station", Letter from Tanya Mensah (NRC) to Biff Bradley (NEI) dated August 28, 2007 (ML072220129)

1. Section I-3.0 of the ASME Code Case N-660, Rev. 0, refers to shutdown, fires, flooding and seismic (hereafter referred to as "external events") as providing information relevant to classification. Although external events are often not modeled in a probabilistic risk assessment (PRA), Tables I-1 to I-4 in ASME Code Case N-660, Rev. 0, may be used to classify structures, systems, and components (SSCs) needed to respond to these external events.

   The proposed methodology[1] retains the original discussion and again mentions external events in a new section (Section I-3.1.2), but provides no additional

[1] Table A-2 of TR WCAP-16308-NP identifies a number of differences between the process described in ASME Code Case N-660 and that applied by Wolf Creek Generating Station (WCGS) and other differences have been identified that are not included in Table A-2. The body of TR WCAP-16308-NP also provides some limited guidance as illustrated here. The process applied by WCGS (including any revision that may be made during the NRC staff review of TR WCAP-16308-NP) is referred to as the proposed methodology.

guidance. The pilot plant did not have an external event PRA and did not use Tables I-1 to I-4.

TR WCAP-16308-NP provides some discussion about external initiating events in the last paragraph on page 4-3 which states:

> "Also, only qualitative risk assessments exist for fire, seismic, external events and shutdown at WCGS. Therefore, to capture the risk importance of piping segments from the fire, seismic, external events and shutdown qualitative risk assessments, any piping segment supporting a high risk significant safe shutdown pathway would be a candidate medium safety significant pipe segment. This is equivalent to the active component classification process where active SSCs that support safe shutdown pathways are not automatically classified as high safety significant, but rather are left to the IDP for a final classification."

The NRC staff believes that the last sentence above is incorrect. As stated at the bottom of page 5, and in the third bullet on the top of page 6, in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," an SSC identified as high safety significant (HSS) by a non-PRA method must remain HSS and may not be reclassified by the Integrated Decision-Making Panel (IDP). The paragraph in TR WCAP-16308-NP places these SSCs into a medium safety significant classification which does allow the IDP to reclassify the SSC as a low safety significant (LSS) SSC.

Please provide a description of how piping segments supporting a safe shutdown pathway that is obtained from a non-PRA analysis of external events should be identified and classified. If the proposed method differs from the method described in NEI 00-04 for active SSCs, please justify this difference.

RESPONSE:

The methodology used at Wolf Creek for the passive classification of pressure boundary components using a non-PRA method was consistent with the NEI 00-04 guidance. Components that support safe shutdown pathways at Wolf Creek were ranked as HSS during the preliminary classification and the Integrated Decisionmaking Panel (IDP) could not re-classify them into a lower risk category. Further, as discussed in Section 3.7 of WCAP-16308-NP[2], the IDP is provided with documentation from the passive categorization and is trained in the roles and responsibilities of the IDP, including passive categorization considerations.

The last sentence in the last paragraph on page 4-3 of WCAP-16308-NP is incorrect. The last sentence of the last paragraph on page 4-3 of WCAP-16308-NP will be revised to read:

> "Also, only qualitative risk assessments exist for fire, seismic, external events and shutdown at WCGS. Therefore, to capture the risk importance of piping segments from the fire, seismic, external events and shutdown qualitative risk assessments, any piping segment supporting a safe shutdown pathway would be classified as a high safety significant pipe segment. This is consistent with the active component classification process where active SSCs that support safe

[2] The IDP described in Section 3.7 of WCAP-16308-NP was used to meet the Code Case N-660 requirement in paragraph 1320 for diverse engineering disciplines in the passive categorization process. This clarification was added to Supplemental Table A-2b.

shutdown pathways are automatically classified as high safety significant and therefore not eligible to be ranked lower by the IDP."

Supplemental Table A-2[3] will be revised to reflect that no change was made to this section from Revision 0 of Code Case N-660 by deleting the line referring to N-660 Section I-3.1.2 from the table.

It is also noted that the consequence assessment from ASME Code Case N-578, "Risk Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1", is the basis for the consequence assessment used for the passive categorization in Code Case N-660, Revision 0. The consequence assessment categorizes piping from an internal events perspective as well as from an external and shutdown events perspective and ranks the piping accordingly. Therefore, the use of Code Case N-660, Revision 0 addresses external events through the consequence assessment.

2. On July 11, 2007, a Category 2 public meeting was held between the NRC staff and industry representatives at NRC headquarters. During the meeting, industry representatives provided a supplemental Table A-2 (that added a large number of entries) to discuss its draft comments in response to the NRC staff's comments on the 50.69 pilot documentation guidance (ADAMS Accession No. ML071930260). As described under the entry for I-3.1.1(a) in the supplemental Table A-2 (but not included in Table A-2 of TR WCAP-16308-NP), the proposed methodology modifies the Section I-3.1.1(a) of ASME Code Case N-660, Rev. 0, to expand the available alternatives to analyzing less that a large pressure boundary failure. The new alternative permitting the analysis of a smaller pressure boundary failure is:

   (4) when design insights do not support a large break based on pressure/temperature/ flow in the pipe segment.

   This guidance provides no predictability about which segments will be assigned a small leakage and which segments would not. Please provide additional guidance that clearly defines the "design insights" and identify criteria that would be used to conclude that the insight does not support a large break. Justify how these insights and criteria provide confidence that a large break in not a credible failure mode.

   RESPONSE:

   Any discussion of break size must start with the identification of the pipe failure mechanisms. The most recent study of pipe breaks is contained in NUREG-1829 "Estimating Loss of Coolant Accident (LOCA)" which was developed through an expert elicitation process to determine the frequency of loss of coolant accidents to support the rulemaking for 10 CFR 50.46a. While the information in that report concentrates on reactor coolant pressure boundary components, a number of insights can be drawn from the information contained in the report, including pipe break morphology, operational experience and probabilistic fracture mechanics assessments.

[3] Supplemental Table A-2 was provided to the NRC during a Category 2 public meeting on July 11, 2007 and archived under ADAMS Accession No. ML071930260.

The consideration of the appropriate break size to be used for considering the consequences of pipe breaks must start with the contributors to potential pipe breaks. As discussed in NUREG-1829, there are five key contributors to pipe breaks: geometry, materials and fabrication, loading history, degradation mechanisms, and mitigation / maintenance. It is noted that there are key interactions between certain considerations for each of the contributors that greatly influence the overall potential for and characteristics of a pipe failure. For example, flow accelerated corrosion (degradation mechanisms) is predominant in carbon steel piping (materials and fabrication), occurs predominantly at elbows and tees in the piping systems (geometry) and is monitored through mitigation measures (mitigation / maintenance) to ensure replacement before pipe failures occur. A detailed discussion of the considerations for each key contributor is provided in NUREG-1829.

A number of key insights can be taken from NUREG-1829:

- Large pipe breaks are highly unlikely without a severe transient load; cracks in piping systems will only leak until they are detected, and then the leak can be repaired.
- Based on operational experience, small pipes are more susceptible to large breaks than large pipes. A given flaw size represents a larger fractional pipe diameter for smaller diameter pipes. Smaller piping is also often subject to fabrication flaws which exacerbates this decreased failure margin. Additionally, smaller diameter lines are often fabricated from socket welded piping which has a history of mechanical fatigue damage from plant vibrations and is also susceptible to external failure mechanisms arising from human error (e.g., damage from equipment). Finally, small piping is typically more difficult to inspect and in-service inspection is not routinely performed on these lines.
- Through-wall flaws that result in leakage make up the majority of the operating history. While this substantiates the leak-before-break philosophy, it also shows the extreme conservatism in the large break assumption. This is apparent for all pipe sizes, from the small vent and drain lines to reactor coolant loop piping.
- Mechanical fatigue was noted to be one of the foremost causes of through-wall flaws, followed by fabrication defect and repair. Those failure mechanisms presently identified through condition monitoring programs (e.g., stress corrosion cracking [SCC] and flow accelerated corrosion [FAC]), are generally not dominant failure mechanisms.

Studies such as that reported in ASME Whitepaper 2002-02B-01, "Alternative Pressure Boundary Treatment Practices for Class 2 and 3 Service Water Systems", have compared the service history of piping designed to different ASME requirements. This study focused on the reliability of raw water systems for plants that have used B31.1 (and AWWA) versus ASME Section III. If the entire operating history is used, plants designed to B31.1 (and AWWA) have a slightly higher piping failure rate than Section III plants. If the operating years prior to 1983, during which many degradation mechanisms were identified, are not considered, then the existing B31.1 (and AWWA) plant piping systems are actually operating more reliably than Section III plants. Others studies referenced in the ASME White Paper have also shown that there is little difference in failure probability for design loading conditions

between the various design codes (e.g. B31.1, ASME Section III), in particular for low temperature systems.

Separate considerations need to be evaluated for seismic loadings on piping segments in determining an applicable break size. Seismic loadings have the potential to cause catastrophic piping failures if the pipe loadings exceed their design basis. There are three important aspects to be considered in seismic evaluations:

- 50.69(d)(2) requires that reasonable confidence be maintained that RISC-3 SSCs can perform their design basis functions under their design basis accident conditions, including seismic and environmental conditions and effects throughout their service life. As discussed in the response to RAI#11, the seismic requirements for repair and replacement of low risk significant components will consider the appropriate seismic conditons to provide that reasonable confidence.
- The resolution of Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46" included performing seismic verifications of certain classes of mechanical and electrical equipment. The resolution methodology proposed by the industry Seismic Qualification Users Group (SQUG) in their "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment" was based on generic equipment earthquake experience data supplemented by generic equipment test data. With respect to pressure boundary components, the SQUG conclusion reflected in the GIP-2 report was that there is adequate seismic capacity for properly anchored equipment in older operating plants. The staff generic Safety Evaluation of the GIP-2 methodology concluded that the GIP-2 approach provides an adequate level of safety and that it was not cost-justifiable for the safety benefit gained to demonstrate the seismic qualification of equipment in these older operating plants by using rigorous current qualification requirements.
- It is also noted in the NUREG-1829 study that small piping using socket welds are susceptible to external failure mechanisms such as seismic loads.

In conclusion, it is proposed that break sizes other than large breaks can be used in the passive categorization subject to the conditions described below:

- A review needs to be conducted to assure the system/segment is not susceptible to any large break mechanisms (as described below) or that plant controls are in place (e.g. condition monitoring) to minimize the potential for occurrence of large break mechanisms. This includes a review of plant and industry operating experience to characterize the potential for piping pressure boundary failure, including unacceptable flaw growth, leaks, failures and degradation processes. Plant specific service history is a key element in identifying degradation mechanism susceptibility because of the uniqueness of particular plant configurations and service conditions to small or large leak applicability.

  A large break mechanism is defined as one that includes significant loadings above the normal loading on the system and specifically includes water hammer for which no mitigation is provided and internal deflagrations, but excludes seismic based on considerations stated above.

- The pipe segment is not part of a high energy system. A high energy system is defined as a system that, for the major operational period, is either in operation or maintained pressurized under conditions where either, or both, of the following are met: a) maximum operating temperature exceeds 200 degrees F, and b) maximum operating pressure exceeds 275 psi.
- The pipe is greater than 4 inches in diameter. This was chosen to coincide with the ASME definition of small bore piping. It also approximates the leak rate for Category 3 for PWRs in NUREG-1829. There is a significant decrease in the likelihood of piping failures between Category 2 and Category 3 for PWRs. This represents a break probability over two orders of magnitude less than the most likely pipe break. This also eliminates concerns about socket weld and support failures of small piping during seismic events.
- The considerations that permit the use of a small break in the passive categorization need to be clearly noted as a basis for the passive categorization to ensure that:
  - Appropriate design and operation measures are maintained to assure that reasonable confidence is maintained so that the plant can perform its design basis function under design basis conditions, considering seismic and environmental conditions, and
  - Post-implementation monitoring and possible corrective actions are based on the appropriate categorization basis.

The appropriate small break size for consideration in passive categorization is the calculated leak rate at normal operating conditions for a through-wall flaw with a length 6 times its depth. This is consistent with the NUREG/CR-4550 definition for a PRA small break LOCA and with the NUREG-1829 results that show this is the most likely leak rate to occur in both PWR and BWR plants. This is also consistent with the operating experience that pipe failures are dominated by mechanical fatigue and fabrication defects, both of which exhibit leak before break characteristics. A flaw aspect ratio of 6 is also commonly used for structural evaluations.

Supplemental Table A-2a will be revised at the entry for I-3.1.1(a) to reflect that smaller break sizes can be considered when certain design and operational considerations can be satisfied by inserting a new item (4) that reads:

"(4) a small break with a calculated leak rate at design basis conditions for a through-wall flaw with a length six times its depth can be used when certain design and operational considerations are satisfied:

- the pipe segment is not susceptible to any large break mechanisms or plant controls are in place to minimize the potential for occurrence of large break mechanisms,
  - a large break mechanism is one that produces significant loadings above the normal loading on the system and specifically includes water hammer for which no mitigation is provided and internal deflagrations, but excludes seismic,
- the pipe segment is not part of a high energy system,

- the pipe segment is greater than 4 inches in diameter."

3. As described under the entry for Section I-3.1.2(b) in Table A-2 of TR WCAP-16308-NP, the NEI proposed new text to be used instead of the text in ASME Code Case N-660, Rev. 0. The single sentence in Section I-3.1.2(a) of ASME Code Case N-660, Rev. 0, is to be expanded into four bullets. It is not clear that the proposed text does not change the original process.

   a. Please identify the Risk-Informed Inservice Inspection (RI-ISI) program criteria (i.e., document and page number) referred to in the explanatory note in this entry in Table A-2.

   b. Please describe each of the proposed changes and provide examples illustrating the differences and similarities between the endorsed ASME Code Case N-660, Rev. 0, text and the proposed text of Section I-3.1.2(b).

   RESPONSE:

   The proposed modification to Section I-3.1.2(b) of Code Case N-660 was not intended to change the process or methodology. The text for Section I-3.1.2(b) in Revision 0 of Code Case N-660 was used for the Wolf Creek categorization.

   Supplemental Table A-2 will be revised to reflect that no change was made to Section I-3.1.2(b) from Revision 0 of Code Case N-660 by deleting the line referring to Section I-3.1.2(b) from the table.

   A proposed modification to Section I-3.1.2(d) of Code Case N-660 was made to be consistent with the proposed modification to Section I-3.1.2(b) of Code Case N-660. The categorization process for Wolf Creek also used the text for Section I-3.1.2(d) of Revision 0 of Code Case N-660.

   Supplemental Table A-2 will be revised to also reflect that no change was made to Section I-3.1.2(d) from Revision 0 Code Case N-660 by deleting the line referring to Section I-3.1.2(c) from the table.

4. During the May 17, 2007, audit of the WCGS IDP documentation, the NRC staff noted that the piping attached to the reactor sump screens was classified as LSS while the screens themselves had been categorized HSS during the active SSC classification phase. After several discussions with industry representatives, it appears that the reactor coolant recirculation function of these screens was not included in the passive classification process because the passive categorization only included the containment spray system functions. At WCGS, failure of the containment spray system does not affect core damage or large early release. Page 27 of NEI 00-04 states, "there may be circumstances where the categorization of a candidate low safety-significant SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system." This caution is not included in the proposed passive categorization methodology.

   a) Please provide additional guidance that provides confidence that piping segments that support two or more systems' functions will be classified based on the highest safety significance function being supported.

b) The proposed method does not appear to require identification and resolution of differences between the safety significance classification between an active SSC and the piping attached to the SSC. Under what conditions is it reasonable for the safety significance of the pressure boundary function of a piping segment to be classified lower than the SSCs to which it is attached?

RESPONSE:

If a piping segment supports more than one function, the piping segment should be classified to the highest safety significance of the functions that it supports. In fact, Section I-3.1.3(a)(3) of the N-660 Code Case, as modified in the response to RAI #5, directs that an assessment of the impact of the failure of a piping system on other systems be undertaken. The intent of this criterion is to ensure that a pipe segment is classified to the highest safety significance of all of the functions that it supports. Using this criterion, the passive categorization of the sump screen should consider both the containment spray recirculation function and the core cooling recirculation function. Since the core cooling recirculation function was not categorized when the containment spray system categorization was undertaken, the containment sump screen classification should not have been completed and its original classification (e.g., high safety significance) should have been retained.

The safety significance of the pressure boundary function of a piping segment should normally be consistent with the safety significance of the active function of a component attached to the subject piping segment, except in certain circumstances. All such circumstances would need to be justified on a case by case basis. Examples of some circumstances are:

- Piping segments are defined based on similarities in consequences. Many times valves are at a boundary between two piping segments and are therefore associated with each piping segment. For example, a containment isolation valve can act a boundary between two piping segments. In this case, the active and passive function ranking of the valve itself would likely be high based on providing a safety significant containment isolation function. However, the associated piping in the piping segments on either side of the isolation valve could be low if they do not serve a fission product release mitigation function. In this case, the active and passive categorization of the isolation valve would be high while the piping segments associated with the valve could be low.
- The failure of a pipe segment may be high because it can result in the draining of a tank that would fail a high safety significant function while an active failure of the pumps taking suction from the tank might be low based on the number of trains in the system or diverse means of performing the high safety significant function. For example, a failure of the piping segment on the suction side of a high head safety injection pump might be high based on draining the suction source for safety injection while the active ranking for that train of safety injection might be low based on multiple trains or diverse means of safety injection such as safety grade charging pumps. This would result in a high passive ranking for the pipe segment but a low ranking for the active function that the pipe segment supports.

To ensure that it is clear that the passive categorization of pipe segment is based on the highest safety significance of all of the functions that it supports, additional

guidance will be included in Section 4.5 of WCAP-16308-NP. A new paragraph will be inserted at the top of page 4-5 to read:

> "Piping segments shall be ranked based on the highest safety significance of all of the functions that it supports. If the importance of all functions that it supports has not been completed, the piping segment must retain its original classification until the importance of all supporting systems has also been evaluated."

To clarify that the passive categorization ranking of a piping segment should generally be consistent with the active categorization for the function that it supports, additional guidance will be included in Section 4.5 of WCAP-16308-NP. Directly following the proposed additional paragraph directly above, another paragraph will be inserted that reads:

> "The safety significance of the pressure boundary function of a piping segment should be consistent with the safety significance of the active function of a piping segment, except in certain circumstances. All such circumstances would need to be justified on a case by case basis."

The guidance in Section 4.5 of WCAP-16308-NP is used during the preliminary engineering passive classification of the pipe segments. As described in Section 3.7 of WCAP-16308-NP, this guidance will also used by the IDP to finalize the pipe segment passive classifications.

5. As described in the entry under Section I-3.1.3(a)(3) in Table A-2 of TR WCAP-16308-NP, the NEI proposed to use new text instead of the text in the endorsed version of N-660. ASME Code Case N-660, Rev. 0 states,

   > "Even when considering operator actions used to mitigate an accident, failure of the piping segment will fail a high-safety-significant function."

   This text has been moved to Section I-3.2.2(b)(1) and modified to now state,

   > "Even when taking credit for plant features and operator actions, failure of the piping segment will not[4] directly fail another high-safety-significant function."

   The introduction of the word "another" in the proposed version significantly alters when the response to this question would be "True" and "False" in a manner which requires further explanation. The original text ensures that a piping segment that would disable any single HSS function would be classified HSS. In the proposed revision, a second (i.e., "another") HSS function would have to be failed in addition to whatever function that the piping segment being classified would directly degrade or fail. Is the intent of this proposed text to require that a second HSS function be consequently failed? If so, please justify not assigning a HSS classification to an SSC whose failure could consequently fail an HSS function.

   RESPONSE:

   The proposed modification was not intended to change the process or methodology in Revision 0 of Code Case N-660. Since no change of process or methodology was

[4] The negative in the proposed methodology is a natural consequence of changing the way "true" and "false" responses are used in the IDP classification as discussed further in .

implemented by Wolf Creek, this section will be returned to the text in Revision 0 of Code Case N-660.

Supplemental Table A-2 will be revised to reflect that no change was made to this criterion by using the wording:

> "Even when taking credit for plant features and operator actions, failure of the piping segment will not directly fail a high safety significant function."

In addition, Consideration 1 on page 4-4 of Section 4.5 of WCAP-16308-NP will be revised to read:

> "Even when taking credit for plant features and operator actions, failure of the piping segment will not directly fail a high safety significant function."

The discussion under Consideration 1 of Section 4.5 of WCAP-16308-NP does not require any change because the process and methodology used by Wolf Creek is consistent with the text in Revision 0 of Code Case N-660.

Also, as noted in footnote 4 on the previous page, the considerations in Section I-3.1.3(a) of Revision 0 of Code Case N-660 were changed in the way that "true" and "false" are used by reversing the responses for a given condition. Human performance fundamentals suggest that the wording of equivalent considerations between the active and passive categorization guidance should be as similar as possible. Therefore, the passive categorization perspective for these considerations was changed to be consistent with the NEI 00-04 considerations. By making this change, the IDP, that makes the final passive and active classification determinations as described in Section 3.7 of WCAP-16308-NP, will be making consistent responses. For an equivalent consideration using the Code Case N-660, Revision 0 guidance, to come to a low safety significance finding the IDP would respond in the positive (e.g., true) for the active ranking and in the negative (e.g., false) for the passive ranking. Therefore, all of the considerations in Section I-3.1.3(a) were revised to have a "true" response for a low safety significance finding from both active and passive categorization. To ensure that this change is highlighted in WCAP-16308-NP, a new bullet will be added in Section 4-5 on page 4-4 to read:

- "All of the considerations in Section I-3.1.3 of Code Case N-660 were changed so that the response (i.e., true or false) for the passive categorization would match the response for the equivalent consideration from the active categorization process in NEI 00-04."

6. As described in the entry under Section I-3.1.3(b)(2) in Table A-2 of TR WCAP-16308-NP, you have proposed to use new text instead of the text in ASME Code Case N-660, Rev. 0. The endorsed version of ASME Code Case N-660 states,

   > "The piping segment supports a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines."

   This text has been moved to Section I-3.2.2(b)(4) and modified to now state,

"The piping segment does not[5] individually support a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines, with no redundancy or alternate means of support."

The introduction of the phrase "with no redundancy or alternative means of support" in the proposed version significantly alters when the response to this question would be "True" and "False" in a manner which requires further explanation. The original question addressed two issues, a particularly important aspect of defense-in-depth and the complexity of modeling human errors. One of the defense-in-depth considerations is to avoid over-reliance on programmatic activities to compensate for weakness in plant design. In this case, relying on the operators to overcome failures which reduce diagnosis information relied upon to mitigate accidents. Quantitative evaluation of the impact of these failures may provide additional information about the impact of these failures on risk and how that impact compares to the acceptance guidelines, but such calculations are very resource intensive and of limited accuracy.

The NRC staff has not yet concluded whether the original statement was too limiting, as argued in TR WCAP-16308-NP, but considers that the introduction of the "individually supports" may provide reasonable flexibility commensurate with the safety significance of the piping. However, because of the pervasive inclusion of instrumentation throughout the plant that normally includes measurements of many related parameters, it would appear that there would never be a piping segment failure for which the response to the proposed question would be "False."

a) Please explain the difference between "individually support" and "no redundancy."

b) Please define "alternative means of support" and justify that full loss of a diagnosis function would not be expected to be safety significant unless these alternative means are also lost. For example, upon loss of the reference leg for level measurement in a refueling water storage tank, would low pressure in the high-pressure safety injection pump inlet (or some other indication) provide an acceptable alternative means for determining when to switch over from injection to recirculation?

RESPONSE:

The Emergency Operating Procedure (EOP) considerations in the passive categorization process used at Wolf Creek were consistent with those used in the active categorization as defined in Section 9.2.2 of NEI 00-04.

In NEI 00-04, Consideration 4 states:

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

Further, Consideration 5 from Section 9.2.2 of NEI 00-04 states:

[5] The negative in the proposed methodology is a natural consequence of changing the way "true" and "false" responses are used in the IDP classification.

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

As discussed in the response to RAI #5, human performance fundamentals suggest that the wording of equivalent considerations between the active and passive categorization guidance should be as similar as possible. By making this change, the IDP, that makes the final passive and active classification determinations as described in Section 3.7 of WCAP-16308-NP, will be making consistent responses.

Therefore, the entry in the third column of Supplemental Table A-2 for I-3.1.3(b)(2) will be revised to read:

"The piping segment is not relied upon to support an active function 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, or for achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions."

In addition, Consideration 4 on page 4-5 of Section 4.5 of WCAP-16308-NP will be revised to read:

"The piping segment is not relied upon to support an active function 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, or for achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions."

The discussion under Consideration 4 in Section 4.5 of WCAP-16308-NP does not require any change because the process and methodology used by Wolf Creek is consistent with the text in Revision 0 of Code Case N-660.

7. The proposed methodology proposes to address the safety significant implication of known active degradation mechanisms using a new question described in the entry under Section I-3.2.2(b)(5) in the Supplemental Table A-2 (ADAMS Accession No. ML071930260). The proposed question states that "the plant condition monitoring program would identify any known active degradation mechanism in the pipe segment prior to its failure in test or actual demand event." The second sentence in Section I-3.2.2(b) in Code Case N-660, stated, "Any piping segment initially determined to be a Medium consequence category and that is subject to a known active degradation mechanism shall be classified HSS." Evidently, the proposed method replaced the guidance in ASME Code Case N-660, Rev. 0, with the guidance under the new Section I-3.2.2(b)(5). This change to ASME Code Case N-660, Rev. 0, will almost certainly result in a number of segments that would have been classified HSS, according to the Code Case, to be classified LSS according to the proposed method.

As written in the proposed methodology, the simple existence of a degradation monitoring program at a plant would seem to result in a "True" designation for every location in the plant that may be susceptible to that degradation mechanism, regardless of whether there are any inspections in the segment being classified. This interpretation is supported by the observation during the NRC staff audit of the WCGS IDP documentation, that the WCGS IDP used the phrase, "[a] plant conditioning monitoring program exists" in a number of places. No other discussions about degradation mechanisms were identified during the audit.

The generic disposition of all known, active degradation mechanisms is contradictory to ASME Code Case N-660, Rev. 0. Please provide additional description about how active degradation mechanisms should be incorporated into the safety-significance classification for a segment. The discussion should describe the relationship between the plant's degradation monitoring programs, the inspection locations within the programs, and the inspection locations within the segment being classified. Please describe the differences between the results that would be obtained using the endorsed code case and the results that will be obtained using the proposed method, and explain why these differences are acceptable.

RESPONSE:

The second sentence in Section I-3.2.2(b) in Revision 0 of Code Case N-660 states:

> "Any piping segment initially determined to be a 'Medium' consequence category and that is subject to a known active degradation mechanism shall be classified HSS."

During the preliminary Wolf Creek passive categorization, it was identified that a significant number of piping systems (and therefore pipe segments) are potentially subject to known active degradation mechanisms, such as flow accelerated corrosion (FAC), microbiologically influenced corrosion (MIC), thermal fatigue, stress corrosion cracking (SCC), etc. Wolf Creek has condition monitoring programs to: a) identify active degradation mechanisms applicable to Wolf Creek, b) identify the piping systems and locations subject to the active degradation mechanisms, c) periodically assess the extent of degradation, and d) initiate corrective actions when necessary to prevent pipe failures due to these degradation mechanisms. Therefore, it is unduly conservative to classify any pipe segment that is subject to active degradation mechanisms as high safety significant without consideration of the assurance provided by the condition monitoring programs.

10 CFR 50.69(d)(2) requires that a licensee implementing 50.69 shall ensure, with reasonable confidence, that RISC–3 SSCs remain capable of performing their safety related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. In addition, 50.69(d)(2) requires that the treatment of RISC–3 SSCs must be consistent with the categorization process and that inspection and testing, and corrective action shall be provided for RISC–3 SSCs. Finally, 50.69(d)(2) and (e) requires that the performance of RISC-3 SSCs be monitored and corrective actions be taken.

The Wolf Creek condition monitoring programs provide reasonable confidence that the piping systems can perform their design basis functions by a multi-step process. First, industry experience and plant experience are periodically reviewed to identify active degradation mechanisms that are applicable to Wolf Creek piping systems.

Second, the condition monitoring programs identify the candidate piping systems and locations that may be subject to the various active degradation mechanisms and an acceptance criterion is developed to provide confidence that a pipe rupture would not be expected. Third, the candidate degradation locations are periodically assessed to determine the extent of degradation that is occurring. The assessment can include periodic monitoring of the degradation through non-destructive processes (e.g., for flow accelerated corrosion), through evaluations (e.g., thermal degradation) or a combination of the two (e.g. stress corrosion cracking). The final step involves the potential for corrective action based on the periodic assessment and monitoring. The corrective action can involve trending assessments, extent of condition assessments, apparent cause assessments, etc. When warranted, the corrective action can include repair and replacement activities.

The Wolf Creek condition monitoring programs are applied to both safety related and non-safety related systems to provide assurance that safety is maintained and that the plant is a highly reliable source of electricity production. Re-classifying piping segments from high safety significance to low safety significance should have no impact on the condition monitoring that is done for those piping systems.

Also it is noted that the passive categorization methodology in Code Case N-660 assumes that a break occurs in a pipe segment with a probability of 1.0. This consideration is applied regardless of whether a potentially active degradation mechanism is present. That is, the impact of the postulated break (e.g. initiating event, number of unaffected systems, and impact on containment) is the same and the resultant consequence rank is the same regardless of the potential for active degradation mechanisms.

Based on the considerations outlined above, the last column of the row identified as I-3.2.2(b) in Supplemental Table A-2a will be changed to read:

> "Continued condition monitoring for known active degradation mechanisms would be a consideration in meeting 50.69 (d)(2) and (e) and therefore classification of HSS is unduly conservative."

No change is required for the row identified as I-3.3.3(b)(5) in Supplemental Table A-2a.

8. Table A-2 provided in TR WCAP-16308-NP is incomplete. Page A-5 states that, "[n]ot all modifications to the code case are reported. Only those differences that could impact the categorization process used a WCGS are shown in Table A-2." The Table did not include a number of differences that have a major impact on the process. During a July 11, 2007, NRC public meeting, Westinghouse representatives provided a supplemental Table A-2 that added a large number of entries. The supplemental Table still does not identify all of the differences between the proposed method and ASME Code Case N-660, Rev. 0.

   For example, the new question listed under Section I-3.2.2(b)(5) in RAI question #7 was not included in Table A-2 in Revision 0 of TR WCAP-16308-NP. The question was included in the supplemental Table A-2. However, in this supplemental table, the entry under "Endorsed Revision 0" was N/A while, in practice, this question replaced the guidance on the same subject that was in Section I-3.2.2(b) of Code Case N-660. There was an entry under I-3.2.2(b) in Table A-2 of TR WCAP-16308-NP but the

entry only refers to the first sentence in Section 3.2.2(b) in the Code Case and stated that "new considerations have been provided."

Not included in either table, nor the TR WCAP-16308-NP, is the deletion of the Code Case's guidance on how degradation mechanisms are to be incorporated into the categorization process. Please submit a table that includes all differences between the endorsed ASME Code Case N-660, Rev. 0, and the proposed method for which approval is being requested. Based on the problems associated with only identifying important differences in the previous tables, please include all differences in the table.

RESPONSE:

Supplemental Table A-2 has been be revised to be consistent with the methodology used by Wolf Creek. Additionally, changes identified in the response to these RAIs have also been incorporated into Supplemental Table A-2. This updated Supplemental Table A-2 is included as a separate attachment to the letter transmitting these RAIs and will replace the current Table A-2 in WCAP-16308-NP.

9. When used in support of the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, a categorization process must include an evaluation that provides reasonable confidence that sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) are small. Please explain how a licensee applying this methodology to categorize the passive SSCs can satisfy 10 CFR 50.69(c)(1)(iv) and provide reasonable confidence that sufficient safety margins are maintained and that any potential increases in CDF and LERF are small.

RESPONSE:

The 50.69 (d)(2) requirement 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 would also apply to low safety significant pipe segments. Therefore, safety margins are not impacted because there are no proposed changes to the plant design basis. Additionally, the monitoring and corrective action required by 50.69(d)(2) assures that any changes in performance of the pressure boundary components would be identified, assessed and corrected as necessary to meet the requirements of the rule. As discussed in the response to RAI #7, this would include known active degradation mechanisms that could impact safety margins of the pressure boundary function.

In addition, the passive categorization process includes a consequence assessment and qualitative considerations to assure that potential changes in CDF and LERF are small. The consequence assessment is applied to all piping segments regardless of whether or not the pressure boundary function is explicitly modeled in the PRA. This provides a first level of confidence that the delta CDF and LERF remain small. The qualitative considerations that are important to risk are applied to the Medium and Low consequence category piping segments to further ensure that the delta CDF and LERF remain small. Finally, 50.69(d)(2) requires that the performance of low safety significant components be monitored and that changes to their performance be assessed in light of the assumptions made in the categorization process. Thus,

any degradation in performance would be assessed to ensure that the delta CDF and LERF remain small.

10. Section 7.3, "Monitoring of RISC-3 SSCs," discusses the review of failures of low-risk safety-related (RISC-3) SSCs as part of the monitoring process under 10 CFR 50.69 of the NRC regulations. Discuss plans to monitor corrective action for degradation of RISC-3 SSCs.

RESPONSE:

Wolf Creek has not developed plant specific methods for monitoring of correction actions to address degradation of RISC-3 SSCs. Note that Wolf Creek uses the term RISC-3 SSC to encompass low safety significant safety related active components classified using the NEI 00-04 guidance as well as low safety significant pressure boundary components classified using the passive categorization process described in WCAP-16308-NP. Plant specific methods, when developed in detail, will follow the approach of NEI 00-04. Specifically, the Wolf Creek monitoring program will include:

- Failures of RISC-3 SSCs will be identified and tracked in a corrective action program.
- Failures of RISC-3 SSCs will be reviewed, as part of the corrective action program, to determine the extent of condition (i.e., whether this failure is indicative of a potential common cause failure).
- Non-failures, such as known active degradation processes, will also be tracked as part of the corrective action program to determine the extent of condition (i.e., the degree of degradation versus the condition monitoring acceptance criterion) and need for corrective actions as discussed in the response to RAI #7.
- Failures of RISC-3 SSCs will be assessed for groups of like component types (e.g., motor operated valves, air operated valves, motor-driven pumps, etc.) for the purposes of assessing data from the corrective action program.
- A periodic review of all failures of RISC-3 SSCs, also considering previous component performance history, will be undertaken at least once every two fuel cycles (per the periodic review schedule recommended in Section 12.1 of NEI-00-04) to:
  - Ensure that the failure rate of RISC-3 SSCs in a given time period has not unacceptably increased due to the changes in treatment. The periodic review will validate that the rate of RISC-3 SSC equipment failures has not increased by a factor greater than that used in the integrated risk sensitivity study.
  - Detect the occurrence of potential inter-system common cause failures, and to allow timely corrective action if necessary.

  If the number of failures for a group of SSCs exceeds the expected number of failures by a factor of two or more, a potential adverse trend is identified requiring further assessment. As a result of the assessment, either:

  - The categorization will be revised to reflect the increased failure rates and the ranking of appropriate SSCs will be reviewed, or

- A corrective action plan will be developed to return the reliability of the SSCs to a level consistent with the categorization.

Section 7.3 of WCAP-16308-NP will be revised to include the response to RAI #10 above.

11. Section 8, "Application of RISC-3 Treatment Requirements," states that the Wolf Creek Nuclear Operating Corporation (WCNOC) will develop and implement documented processes to control the design, procurement, inspection, and maintenance to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Section 8 also states that the WCNOC approach to inspection, testing, and corrective actions is described in Section 7 of the TR. However, Section 7 discusses monitoring of failure rates. Discuss the plans for inspection, testing, and corrective actions for RISC-3 SSCs that satisfy 10 CFR 50.69(c)(1)(iv), (d)(2), and (e). For example, the South Texas Project nuclear power plant is implementing a specific plan for treatment of low-risk safety-related SSCs as part of an exemption received from special treatment requirements in 10 CFR Part 50.

RESPONSE:

Wolf Creek has not developed plant specific methods for inspection testing and corrective actions for RISC-3 SSCs to ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Note that Wolf Creek uses the term RISC-3 SSC to encompass low safety significant safety related active components classified using the NEI 00-04 guidance as well as low safety significant pressure boundary components classified using the passive categorization process described WCAP-16308-NP.

10 CFR 50.69(d)(2) requires that two elements of safety be maintained:

- Reasonable confidence be maintained that RISC-3 SSCs can perform their design basis functions under their design basis accident conditions, including seismic and environmental conditions and effects throughout their service life, and
- The basis for the categorization of RISC-3 SSCs be validated through monitoring of the performance of RISC-3 SSCs and corrective actions be implemented when the categorization basis is not maintained.

10 CFR 50.69(e) requires that changes to the plant, operational practices, applicable plant and industry operational experience be periodically reviewed and, as appropriate, the PRA and SSC categorization and treatment processes be updated.

To comply with the requirements of 50.69(d)(2), and (e), WCNOC will:

- Procure RISC-3 SSCs in a manner consistent with current practices for commercial grade equipment that includes, as a minimum: a) development of procurement specifications that ensure that the component can perform its design basis function under the appropriate design basis conditions, including seismic and environmental conditions and effects throughout their service life,

and b) inspect the equipment upon receipt at the plant to ensure that the proper component was received.

- Periodically maintain and test RISC-3 SSCs in a manner consistent with current practices for commercial grade equipment that includes, as a minimum, development of preventive maintenance requirements and schedules.
- Track and assess failures of RISC-3 SSCs through the corrective action program that includes those actions outlined in the response to RAI #10.

Section 8 of WCAP-16308-NP will be revised to include the response to this RAI and will refer back to Section 7.3 for the discussion of monitoring and corrective actions.

12. Section I-3.1.1(e) of CC N-660 provides for "[p]ossible automatic and operator actions to prevent loss of system function," to be included in the consequence evaluation. WCAP-16308[-NP] indicates that this section is unchanged in its proposed methodology. Please describe how these automatic and operator actions should be included in the consequence evaluation.

RESPONSE:

Section I-3.1.1 refers to the Failure Modes and Effects Analysis (FMEA) that identifies potential failure modes for each piping segment and their effects. The FMEA is then used as input to the Impact Group Assessment described in Section I-3.3.2. The consequence assessment described in Sections I-3.3.1 and I-3.3.2 of Code Case N-660 is taken from Code Case N-578, "Risk Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1". Details of the consequence assessment for Code Case N-578 are documented in EPRI TR-112657, Rev B-A, "Risk-Informed Inservice Inspection Evaluation Procedure" [ADAMS Accession No. ML013470102].

A White Paper is prepared for each ASME Code Case that describes the background for the considerations in the Code Case. The White Paper for Code Case N-660, Revision 0 describes the use of operator actions in the consequence assessment, consistent with the Code Case N-578 process and TR-112657, Rev B-A. In Section 3.3.1, "Fundamental Principles" of TR-112657, Rev B-A, it is stated that:

> "The possibility of isolating a break is also identified and accounted for as part of the consequence analysis. A break could be isolated by a protective check valve, a closed isolation valve, or it could be automatically isolated by an isolation valve that closes on a given signal. If not automatically isolated, a break can be isolated by an operator action, given successful diagnosis. The likelihood of isolating a break depends on the availability of isolation equipment, a means of detecting the break, the amount of time available to prevent specific consequences (e.g., flooding of the room or draining of the tank), and human performance. If isolation is possible, the consequence assessment should be conducted for both cases: successful and unsuccessful isolation. Operator recovery actions are further discussed in Section 3.3.3.2 [of TR-112657, Rev B-A]."

At Section 3.3.3.2.2, "Number of Backup Systems and/or Trains Available" of TR-112657, Rev B-A, under the subheading of "Human Actions as Backup Trains", additional details of the consideration of operator actions in the consequence analysis are provided:

> "Human actions, included in the PSA success criteria, are also credited as backup trains, based on human error probability (HEP). One example is shown in Figure 3-3, where the operator action to initiate feed and bleed is credited in the heat removal function. In addition to human actions modeled in the PSA, the actions to recover from pipe failures, and minimize consequences by isolating breaks, are also modeled in this approach and credited as backup trains. If isolation is possible, consequences should be analyzed for both cases: successful and unsuccessful isolation. In the case where isolation is successful, then the recovered trains or systems are credited. In the case where isolation failed, then, in addition to the isolation, only the remaining trains (if any) are credited. If an unisolated failure would disable all backup trains, the only protection available is isolation of the break.
>
> Operator recovery actions (isolation of the break) can only be credited if:
>
> - There is an alarm and/or clear indication, to which the operator will respond,
> - The response is directed by procedure,
> - The isolation equipment (e.g. valves) is not affected by the break,
> - There is enough time to perform isolation and reduce consequences.
>
> If all of the above factors are satisfied, and can be documented, it is recommended crediting the recovery action, and assuming one backup train "worth" (HEP of approximately 1E-2). The licensee shall not take credit for more than what the recoverable train or system is worth. Additional recovery may be credited on a plant specific basis and should be documented. As necessary, the performance of detailed HRA analysis can be required. Of course, it is left to the analyst to evaluate how reasonable the simplified assumption is and, if necessary, perform a full HEP analysis. It should be noted that, in the addition to the new recovery actions specifically introduced in this analysis, recovery actions already modeled in the PSA can be affected by the analyzed events and need to be reevaluated. If a failure of the system or train is a result of a pressure boundary failure, then the recoveries usually credited in PSA, for example a recovery of the pump, can not be credited."

The following provides an example of the consideration of operator actions in the Wolf Creek categorization of the containment spray system (EN system). The FMEA identified that a number of pipe segments have the potential for the loss of RWST if a failure in an EN pipe segment is not isolated. It was determined that the failure of an EN pipe segment would not result in an initiating event (although it may result in a plant shutdown due to the Tech Specs requirements for a low RWST level). In the System Impact Group Assessment a medium safety significance was assigned to these segments based on the potential loss of RWST if a failure in the pipe segment is not isolated for an accident in which the containment spray system is actuated based on high containment pressure. The medium classification was based on Table I-2 of Code Case N-660 with:

- An "unexpected" frequency of challenge,
- A yearly exposure time, and
- With operator action to isolate the affected EN system train
  - One train of EN still operating for containment cooling; no credit was taken for the containment fan coolers.
- Without operator action to isolate the affected EN system train
  - Credit was taken for the two trains of containment fan coolers for containment cooling.

The potential for success of the operator action to diagnose and take actions to isolate the break was performed for EN pipe segment breaks consistent with the process described in the White Paper.

Consideration of the impact of a loss of the RWST inventory on the emergency core cooling function is assessed using the considerations in Section I-3.1.3 as modified for the Wolf Creek categorization to require use of Section I-3.1.3 for all pipe segments classifed as medium or low from the consequence assessment in Section I-3.1.2 of N-660, Revision 0. In this case, consideration of operator actions to isolate a break in an EN piping segment to preserve RWST inventory for emergency core cooling was applied consistent with the considerations outlined in the TR-112657, Rev B-A and Supplemental Table A-2 entries for new Section 3.2.2(b)(1) and the associated footnote.

Enclosure 2

Table A-2a	Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
-1200(a)	"... failure potential is conservatively assumed to be 1.0 in determining a consequence category in Appendix I."	"...failure potential is conservatively assumed to be 1.0 in performing the initial consequence evaluation per I-3.1 in Appendix I."	To be clear that the failure potential is conservatively assumed to be 1.0 in I-3.1, Consequence Evaluation. This allows the expert panel to assume other than 1.0 for the failure potential when considering the other relevant information in I-3.2 for piping segments determined to be Medium, Low, or None consequence category in I-3.1.
-1200(b)	"Class 1 items that are part of the reactor coolant pressure boundary..."	"items optionally classified to Class 1 and Class 1 items..."	Although this section was modified for the WCGS IDP Version there were no Class 1 items in the two systems evaluated at Wolf Creek. Therefore, this provision was not applied at Wolf Creek. Nonetheless, it was decided that for all future applications at Wolf Creek all Class 1 items will be classified as HSS per the NRC endorsement of N-660 in Reg Guide 1.147, Rev 14.
I-1.0	N/A	Added figure[1] illustrating the modified RISC methodology process, including scope identification, consequence evaluation, consequence categorization, classification considerations, and final classification definitions.	Figure added to provide high level overview of RISC methodology process. New process calls for all segments to be included in the consequence evaluation to determine high, medium, low or none consequence category. Then only the non-high category segments would be considered in the classification considerations of I-3.2.2(b) – previously I-3.1.3.

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Table A-2a Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	**Endorsed Revision 0**	**WCGS IDP Version**	**Basis for Change**
I-2.0	N/A	"Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply."	Although this section was modified for the WCGS IDP Version there were no Class 1 items in the two systems evaluated at Wolf Creek. Therefore, this provision was not applied at Wolf Creek. Nonetheless, it was decided that for all future applications at Wolf Creek all Class 1 items will be classified as HSS per the NRC endorsement of N-660 in Reg Guide 1.147, Rev 14.
I-3.0, Title	"Consequence Assessment"	"Evaluation of Risk Informed Safety Classifications"	For clarification to meet Figure I-1.
I-3.0	"Piping segments can be grouped based on common conditional consequence…"	"All pressure retaining items, including supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common conditional consequence…"	For clarification of the scope of components to be evaluated.
I-3.0	"Additionally, information shall be collected for each piping segment that is not modeled in the PRA, but considered relevant to the classification (e.g., information regarding design basis accidents, shutdown risk, containment isolation, flooding, fires, seismic conditions)."	"Additionally, information considered relevant to the classification shall be collected for each piping segment (e.g., information regarding design basis accidents, at-power risk, shutdown risk, containment isolation, flooding, fires, seismic conditions, etc.). This other relevant information is considered in conjunction with the Consequence Category to determine the Risk Informed Safety Classification."	Statement clarified for other relevant considerations besides internal events PRA.

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Table A-2a	Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	**Endorsed Revision 0**	**WCGS IDP Version**	**Basis for Change**
I-3.1.1(a)	N/A	"(4) a small break with a calculated leak rate at design basis conditions for a through-wall flaw with a length six times its depth can be used when certain design and operational considerations are satisfied: - the pipe segment is not susceptible to any large break mechanisms or plant controls are in place to minimize the potential for occurrence of large break mechanisms, + a large break mechanism is one that produces significant loadings above the normal loading on the system and specifically includes water hammer for which no mitigation is provided and internal deflagrations, but excludes seismic. - the pipe segment is not part of a high energy system, - the pipe segment is greater than 4 inches in diameter."	Consideration given to specific design and operational characteristics of the pressure retaining and support items that can affect the size of failure of the pipe segments.

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Table A-2a Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.1.3, 3.1.4, & 3.1.5	All	Sections has been modified and moved into new section I-3.2.2(b). The process used at the WCGS IDP calls for all segments to be created and assigned a consequence category in Sections I-3.1.1 & 3.1.2. Then, for those segments with a consequence category of MEDIUM, LOW, or NONE, the user must evaluate a modified Sections I-3.1.3, 3.1.4, and 3.1.5 (now in I-3.2.2(b)) to assign final high or low safety significance.	Original intent of section was to provide additional considerations for segments not modeled in the PRA. However, the grouping of components into piping segments and the use of surrogate components in the PRA provide quantitative evaluations for each piping segment. The intent of this section now is to provide further considerations for piping segments with MEDIUM, LOW, or NONE consequence categories. See the following entries for specific changes to the original considerations of I-3.1.3, 3.1.4, and 3.1.5.
I-3.1.3	All	Questions changed such that all TRUE responses will support LSS and at least one FALSE response will support HSS.	For consistency with NEI 00-04 process.
I-3.1.3(a)(1)	"Failure of the piping segment will significantly increase the frequency of an initiating event, including those initiating events originally screened out in the PRA, such that the CDF or large early release frequency (LERF) would be estimated to increase by more than 10-6/yr or 10-7/yr, respectively."	Deleted	Redundant to the considerations in I-3.1.1 and I-3.1.2 when determining failure consequences and consequence category.
I-3.1.3(a)(2)	"Failure of the piping segment will compromise the integrity of the reactor coolant pressure boundary as defined in –1200(b)."	Deleted	All reactor coolant pressure boundary segments are ranked high safety significant per -1200(b).

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Table A-2a	Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.2.2(b)	All	Rather than referring to Sections I-3.1.3, I-3.1.4, and I-3.1.5, new considerations have been provided as listed above. Process still requires user to evaluate the additional considerations for any segment with consequence category Medium, Low, or None.	To improve the process, the additional considerations were moved into this section from I-3.1.3, I-3.1.4, and I-3.1.5. See above for basis of consideration changes.
I-3.2.2(b)	"Any piping segment initially determined to be a Medium consequence category and that is subject to a known active degradation mechanism shall be classified HSS."	Deleted	Continued condition monitoring for known active degradation mechanisms would be a consideration in meeting 50.69 (d)(2) and (e) and therefore classification of HSS is unduly conservative.
I-3.2.2(b)(5)	N/A	"The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program)."	In response to previous change immediately above.

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Table A-2a	Methodology/Process Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	**Endorsed Revision 0**	**WCGS IDP Version**	**Basis for Change**
I-3.2.2(b)	N/A	Following the new 11 considerations, there was a provision added to allow a pipe segment to be ranked as LSS even if one of the 11 considerations was answered FALSE. The provision states: "If any of the above eleven (11) conditions are not true, HSS should be assigned unless the following can be met: • A condition monitoring program would identify the degradation of the piping segment prior to its failure in test or an actual demand event, or • Historical data show that these failure modes are unlikely to occur and such failure modes can be detected in a timely fashion. Historical data should be restricted to items procured to a specification no more stringent than the minimum specification that could be imposed on a similar item determined to be LSS by this process.	This provision was not used at Wolf Creek and will not be used for future Wolf Creek applications. It was also suggested to ASME that this provision be removed from future revisions of N-660.

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Table A-2b	Clarification Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	**Endorsed Revision 0**	**WCGS IDP Version**	**Basis for Change**
Applicability	"... through 2001 Edition"	"... through 2003 Edition"	Updated to be current at the time of the WCGS IDP.
-1200(b)	Entire paragraph	Reworded for clarity	Clarification of the scope of items to be evaluated.
-1320	Entire paragraph	"(a) An Integrated Decisionmaking Panel (IDP) shall use the information and insights compiled in the initial categorization process and combine that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of functions/SSCs. (b) The designated as members of the IDP shall have joint xpertise in the following fields: - Plant Operations (SRO qualified). - Design Engineering. - Safety analysis. - Systems Engineering, and Probabilistic Risk Assessment. (c) Requirements for ensuing adequate expertise levels and training of IDP members in the categorization process shall be established. (d) To the extent possible, the classification of pressure retaining and support items in a system should be performed by the same IDP members as the categorization of active SSCs in that system."	Clarification of the process used for the WCGS categorization of pressure retaining and support items. An initial categorization of pressure retaining and support items was performed by an engineering function. The IDP, composed of the members with expertise in the disciplines identified in the original paragraph -1320, then considered the initial categorization, along with other information from their respective disciplines, to finalize the categorization. The method used at WCGS results in a categorization processes for classifying pressure retaining and support items that is similar to that used for active SSCs. This helps to ensure consistent consideration of information used the two categorization processes.

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Table A-2b Clarification Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
-9000, high-safety-significant function	N/A	Added to end of definition – "or from other relevant information (e.g., defense in depth considerations)"	Added to consider defense in depth in determining the safety significance of a function.
-9000, plant features	N/A	"Plant features – systems, structures, and components that can be used to prevent or mitigate an accident"	Plant features terminology added to Code Case relative to operator and possible automatic actions
-9000, PRA	"a qualitative and quantitative assessment…"	"an assessment…"	Changed to be consistent with the ASME PRA Standard.
-9000, spatial effects	"A failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement or flooding."	"A failure consequence affecting other systems or components, such as failures due to pipe whip, jet impingement, jet spray, loss of inventory due to draining of a tank or flooding."	Including other possible forms of spatial effects.
I-2.0	"The owner shall define the boundaries included in the scope of the RISC evaluation process."	"The owner shall define the boundaries included in the scope of the RISC evaluation process. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class 1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply."	The third and fourth sentences added for clarification of the scope of items to be evaluated. As previously stated, there is no intention for Wolf Creek to rank Class 1 items anything other than high safety significant. The second sentence will not be suggested for future inclusion in N-660.
I-3.0, Title	"Consequence Assessment"	"Evaluation of Risk Informed Safety Classifications"	For clarification to meet Figure I-1.
I-3.0, 1st Paragraph	"Piping segments can be grouped based on common conditional consequence…"	"All pressure retaining items, including supports for a piping system, shall be evaluated by defining piping segments that are grouped based on common conditional consequence…"	For clarification of the scope of components to be evaluated.

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Table A-2b	Clarification Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.0, 1st Paragraph	"Additionally, information shall be collected for each piping segment that is not modeled in the PRA, but considered relevant to the classification…"	"Additionally, information considered relevant to the classification shall be collected for each piping segment…"	Clarifies requirement to collect relevant information for ALL piping segments, not just those modeled in the PRA.
I-3.1.1, 1st Sentence	"Potential failure modes for each piping segment shall be identified…"	"Potential failure modes for each system or piping segment shall be identified…"	Clarify that evaluation should consider system level failure modes as well as piping segment failure modes.
I-3.1.1(c), Indirect Effects	"These include spatial interactions such as pipe whip, jet spray, and loss of inventory effects (e.g., draining of a tank)."	"A failure consequence affecting other systems or components, such as spatial effects."	To be consistent with glossary term for spatial effect.
I-3.1.1(d), Initiating Events	"These are identified using a list of initiating events from any existing plant specific Probabilistic Risk Assessment (PRA) or Individual Plant Examination (IPE) and the Owner's Requirements."	"For systems or piping segments that are modeled either explicitly or implicitly in any existing plant-specific Probabilistic Risk Assessment (PRA), any applicable initiating event is identified using a list of initiating events from that PRA."	Clarify source of initiating events.
I-3.1.2, 3rd sentence	"… (high, medium, low)…"	"… (high, medium, low, or none)…"	"None" is one of the four consequence categories which can be assigned in I-3.1.
I-3.1.2(a)(1)	"The initiating event shall be placed in one of the categories in Table I-1."	"The initiating event shall be placed in one of the Design Basis Event Categories in Table I-1."	More clearly defined what "category" means relative to Table I-1.
I-3.1.2(a)(1)	"… updated final safety analysis report, PRA, or IPE shall be included"	"… updated final safety analysis report or PRA shall be included"	Removed IPE because it was felt that the IPE does not provide any additional information in this area.

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Table A-2b	Clarification Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	**Endorsed Revision 0**	**WCGS IDP Version**	**Basis for Change**
I-3.1.2(b)	"The consequence category of a failure that does not cause an initiating event, but degrades or fails a system essential to prevention of core damage shall be based on the following:"	"The consequence category of a failure: • modeled in a PRA that degrades or fails a high-safety-significant function but does not cause an initiating event, or • not modeled explicitly or implicitly in a PRA, or • that results in failure of another high-safety-significant piping segment, e.g., through indirect effects, or • that will prevent or adversely affect the plant's capability to reach or maintain safe shutdown condition, shall be based on the following:"	Clarified to include the consideration of other consequences of a failure.
I-3.1.2(b)(1)	"Frequency of challenge that determines how often the mitigating function of the system is called upon. This corresponds to the frequency of initiating events that require the system operation."	"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."	Clarified to include functions other than simply mitigating functions and all events as opposed to only initiating events.
I-3.1.2(b)(3)	"Exposure time shall be obtained from Technical Specification limits."	Deleted	Deletion made because it was redundant to the 2nd sentence.
I-3.1.2(b)(3)	"In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5."	Moved out from (b)(3) to directly under (b) and changed text to. "For failures modeled in a PRA, quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-2."	Clarification; this statement applies to all of (b) and not only (3) for Exposure Time.
I-3.1.2(c)	"In lieu of Table I-3, quantitative indices may be used to assign consequence categories in accordance with Table I-5."	"For failures modeled in a PRA, quantitative indices may be used to assign consequence categories in accordance with Table I-5 in lieu of Table I-3. The quantitative index for the combination impact group is the product of the change in conditional core damage frequency (CDF) and the exposure time."	Clarification of the use of Table I-5 and how the combination impact group quantitative index is calculated.

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Table A-2b Clarification Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.1.2(d)	"The above evaluations determine failure importance relative to core damage."	"The above evaluations determine failure importance relative to core damage or the plant's capability to reach or maintain safe shutdown conditions."	Added consistent with the changes made to I-3.1.2(b).
I-3.1.3(a)(3)	"Even when considering operator actions used to mitigate an accident, failure of the piping segment will fail a high safety significant function."	New Section I-3.2.2(b)(1). "Even when taking credit for plant features and operator actions, failure of the piping segment will not directly fail a high safety-significant function."	Added plant features along with operator actions. Footnote provided for credible operator actions (see below).
I-3.1.3(a)(4)	"Failure of the piping segment will result in failure of other safety-significant piping segments, e.g., through indirect effects."	New Section I-3.2.2(b)(2). "Failure of the piping segment will not result in failure of another high safety-significant piping segment, e.g., through indirect effects."	Minor change.
I-3.1.3(a)(5)	"Failure of the piping segment will prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions."	New Section I-3.2.2(b)(3). Even when taking credit for plant features and operator actions, failure of the piping segment will not prevent or adversely affect the plant's capability to reach or maintain safe shutdown conditions.	WCGS IDP was given ability to credit valid operator action when evaluating failure impact on shutdown conditions. Footnote provided for credible operator actions (see below).
I-3.1.3(b)(1)	"The piping segment is a part of a system that acts as a barrier to fission product release during severe accidents."	Deleted	This statement was too conservative to force all segments to be ranked as HSS given that just one segment in the entire system meets this criterion. Also, the intent of this consideration is expressed in new subsections I-3.2.2(b)(6) and (11).

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Table A-2b	Clarification Changes in ASME Code Case N-660 for WCGS Categorization		
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.1.3(b)(2)	"The piping segment supports a significant mitigating or diagnosis function addressed in the Emergency Operating Procedures or the Severe Accident Management Guidelines."	New Section I-3.2.2(b)(4). "The piping segment is not relied upon to support an active function 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 of for achieving actions for assuring long term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities. This also applies to instrumentation and other equipment associated with the required actions."	The original statement was too limiting to any segment supporting functions addressed in the EOPs or SAMGs. The term significant was too vague. New statement is consistent with NEI 00-04 and clarifies the interpretation for the WCGS IDP. It allows for reasonable consideration of plant features and operator actions.
I-3.1.3(b)(3)	"Failure of the piping segment will result in unintentional releases of radioactive material in excess of plant offsite dose limits specified in 10 CFR Part 100."	New Section I-3.2.2(b)(6). "Even when taking credit for plant features and operator actions, failure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions."	The off-site emergency response and protective actions limits are more conservative compared to those in Part 100.
I-3.1.4	All	No change to methodology but the appropriate items called out in Reg Guide 1.174 were placed in I-3.2.2(7) through (11) (see below).	For clarity and process improvement.
I-3.1.5	All	No change to methodology but section was moved to I-3.2.2(c). Format change also made to paragraph to more clearly identify questions for consideration.	For clarity and process improvement.
I-3.2	N/A	Added as first sentence, "Risk Informed Safety Classification is determined by considering the Consequence Category in conjunction with other relevant information."	Added to clarify intent of I-3.2.

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Table A-2b Clarification Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.2.2(b)	N/A	"The following conditions shall be evaluated and answered true or not true:"	Clarification provided to answering the additional considerations as true or not true. If any of the eleven considerations are not true then the segment shall be assigned HSS, otherwise it can be assigned LSS.
I-3.2.2(b), footnote	N/A	To credit operator actions, the following criteria must be met: • There must be an alarm or clear indication of the failure. • A procedure must direct the response to the alarm or indication. • Equipment activated to alleviate the condition must not be affected by the failure. • There must be sufficient time to perform the compensatory action.	Words paraphrased from Supplement 2, Rev 1 of WCAP-14572, Rev 1 – the Pressurized Water Reactor Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications. The guidance is provided for expert panel members when relying on operator actions to make decisions regarding safety significance.
I-3.2.2(b)(7)	N/A	"A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation."	Taken from Reg Guide 1.174.
I-3.2.2(b)(8)	N/A	"Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided."	Taken from Reg Guide 1.174.
I-3.2.2(b)(9)	N/A	"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)."	Taken from Reg Guide 1.174.

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Table A-2b Clarification Changes in ASME Code Case N-660 for WCGS Categorization			
N-660 Section	Endorsed Revision 0	WCGS IDP Version	Basis for Change
I-3.2.2(b)(10)	N/A	"Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed."	Taken from Reg Guide 1.174.
I-3.2.2(b)(11)	N/A	"Independence of fission-product barriers is not degraded."	Taken from Reg Guide 1.174.
I-3.2.2(c)	All	The original text was combined in I-3.2.2(b). The new I-3.2.2(c) is a copy of the original I-3.1.5 section for safety margin assessment.	For simplification and process improvement.
I-3.2.2	A component support or snubber shall have the same classification as the highest-ranked piping segment within the piping analytical model in which the support is included. The Owner may further refine the classification ranking by more extensive application of the process defined in these requirements. These analyses shall be documented.	Moved into I-3.2.2(d) with no change to text.	For consistency.

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Note 1 – Figure I-1, Risk-Informed Safety Classification Process

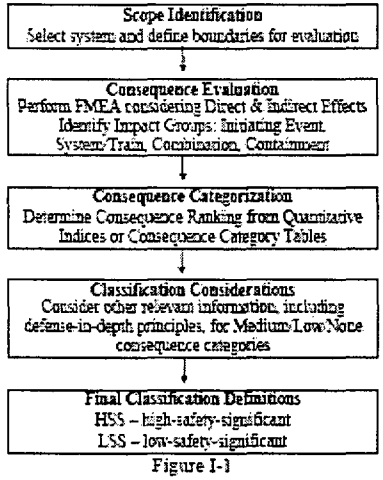

Figure I-1
Risk-Informed Safety Classification Process

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