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 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: It is agreed that the sentence in question from the WCAP-16308-NP is incorrect. The methodology used at Wolf Creek for classification using a non-PRA method was consistent with that stated in NEI 00-04. Components that support safe shutdown pathways at Wolf Creek were ranked as HSS and not given the opportunity to be classified into a lower risk category by the IDP. The WCAP will be revised to read:

[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.

> "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 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, therefore not eligible to be ranked lower by the IDP."

Supplemental Table A-2[2] will be revised to reflect that no change was made to this section from the endorsed version of N-660.

It is also noted that the WCNOC RI-ISI consequence assessment, which is the basis for the consequence assessment used for the passive categorization described in WCAP-16308, categorizes piping from an external and shutdown events perspective and ranks the piping accordingly

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: The NRC convened an expert elicitation for determining the frequency of loss of coolant accidents to support the rulemaking for 10 CFR 50.46a. The results of the expert elicitation are documented in NUREG-1829, Estimating Loss of Coolant Accident (LOCA). While the information in this report concentrates on reactor coolant pressure boundary components, a number of insights can be drawn from the information contained in the report, including the operational experience and the probabilistic fracture mechanics assessments.

[2] 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 operational experience includes piping failures in the older PWRs where the reactor coolant pressure boundary piping designed to the B31.1 code requirements. The B31.1 code requirements might also be appropriate for repair and replacement of components found to be low safety significant from the 50.69 categorization.
- 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 through 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, SCC and FAC), are generally not the dominant failure mechanisms.
- Small leakages, in the range of 100 gpm, are the most probable leakage for all ranges of pipe sizes. The probability of leakages that would result from a large break (e.g., guillotine rupture) of the piping is typically in the E-08 to E-09 per year range.

Based on the information in NUREG-1829, which is consistent with the older NRC studies in NUREG/CR-5750 and 4550, the following quantifiable criteria for break size is proposed:

> The appropriate 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. Additionally, the probability of this leak rate occurring is closest to but still conservative relative to the assumed value of 1.0. This proposed criterion is both quantifiable and predictable.

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 was not intended to change the process or methodology in the endorsed version of N-660. Since no change of process or methodology was implemented by Wolf Creek, this section will be returned to its original text as stated in the endorsed version of N-660. Section I-3.1.2(d) will also be returned to its original text as stated in the endorsed version of N-660 for the same reason.

Supplemental Table A-2 will be revised to reflect that no change was made to this section from the endorsed version of N-660.

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: (a) It is agreed that a given piping segment should be classified to its highest function if it in fact supports more than one function. Also, such guidance should consider that one or more functions could be covered by other systems not included in a given assessment. The ranking of the piping segment, in this case, may be ranked based on it' relative importance to the system being evaluated, but must retain its original classification until the importance of all supporting systems can also be evaluated. Additional guidance will be inserted as bullet items into Section 4.5 of WCAP-16308-NP to read:

- Piping segments shall be ranked based on the highest function it supports.
- The ranking of the piping segment may be ranked based on its relative importance to the system being evaluated but must retain its original classification until the importance of all supporting systems can also be evaluated.

(b) 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 special cases. Such a special case would be a containment isolation valve that acts a pressure boundary between two piping segments; in this case, the active and passive function ranking of the valve would be high while the associated piping in the piping segments could be low. There will also be examples where the active function ranking is low while the passive function may be high (e.g. draining of a tank will fail a whole system while an active failure may fail only one pump train in a three train system). All special cases would need to be justified on a case by case basis.

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[3] 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 the endorsed version of N-660. Since no change of process

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

or methodology was implemented by Wolf Creek, this section will be returned to its original text as stated in the endorsed version of N-660.

Supplemental Table A-2[1] will be revised to reflect that no change was made to this section from the endorsed version of N-660.

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[4] 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 diagnoses function would not be expected to be safety significant unless these

[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.

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 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. To ensure continued consistency between the active and passive ranking considerations Supplemental Table A-2 will be revised to include questions #4 & #5 from Section 9.2.2 of NEI 00-04.

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-.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: In response to RAI #2, the basis for the identification of break sizes for use in this categorization process was provided. These break sizes are then used in the consequence assessment and are postulated with a probability of 1.0. [Note: As discussed in RAI #2, the basis for the break sizes used in the categorization does not discriminate on condition monitoring and concludes that break sizes larger than the recommended break size are extremely low frequency that they are not pertinent to a risk-informed application.] Given these breaks are postulated with a probability of 1.0, whether a potentially active degradation mechanism is present or not, is not relevant. That is, the impact of the postulated break (e.g. initiating event, number of unaffected systems, impact on containment) is the same and the resultant consequence rank is the same.

Within the context of the 50.69 process, a licensee is required to show that there is reasonable confidence that any SSC categorized as LSS will be able to reliably perform its design basis function(s) under accident conditions. Also, the rule requires that the treatment applied to LSS SSCs needs to be consistent with the categorization methods and assumptions.

From a practical perspective, there are many piping systems and locations within the plant that may be subject to known degradation mechanisms such as (but not limited to) FAC, MIC and SCC. Condition monitoring programs are in-place at Wolf Creek to identify any potential piping degradation and take corrective action before a failure occurs. In addition to addressing potential safety concerns, these condition monitoring programs are part of the plant asset management program and therefore have a two-fold purpose.

Given the above, it is proposed to delete this requirement (known active degradation mechanism) from the categorization process.

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.

> Supplemental Table A-2 will be revised to be consistent with the methodology used by Wolf Creek. Additionally, any changes identified to the response to these RAIs will also be incorporated into Table A-2. This revised Table A-2 will be incorporated into a revision of 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: Similar to the active categorization, safety margins are not impacted because there are no proposed changes to the plant design basis. The categorization process, which includes a consequence assessment and qualitative considerations, assures that 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. Additionally, 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, a licensee is required to show that there is reasonable confidence that any SSC categorized as LSS will be able to reliably perform its design basis function(s) under accident conditions thereby providing an additional level of confidence that the delta CDF and LERF are 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. When

the plant specific methods are developed, they will follow the approach of NEI 00-04.

- 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).
- 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 annually 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.

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.

In approving 50.69 for final publication in 2004, the Commissioners expressed the opinion in VR-SECY-04-0109 (Attachment 1) that this is a performance based rule and that treatment requirements for RISC-3 SSCs needed to be less than those for high safety significant SSCs. The Commissioners ordered specific changes to the final rule to ensure that licensees implementing the rule could, in fact, realize an improvement in safety by focusing on risk significant SSCs. To that end, the only "treatment requirements" in the rule are performance based and require 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, 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.

To comply with the requirements of the 50.69 rule 10 CFR 50.69(c)(1)(iv), (d)(2), and (e) as clarified by the Commissioner's comments in VR-SECY-04-0109, 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, 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.

Summary of the Proposed RAI Responses for WCAP-16308-NP, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station

Bob Lutz, Westinghouse Mo Dingler, Wolf Creek / PWROG

NRC Public Meeting on 10 CFR 50.69 Pilot Program

28 September 2007

Westinghouse

Overall Considerations

WCAP-16308-NP provides a methodology for implementing the 50.69 rule exclusive of the PRA technical adequacy

- The methodology uses the categorization that was performed by the PWROG at Wolf creek for the purposes of illustration
- In general, the methodology follows NEI 00-04 and the ASME N-660 Code Case, Revision 0
  - In some instances, exceptions were taken from the categorization processes that have been endorsed by NRC; these exceptions are clearly identified in WCAP-16308-NP
- The PWROG is requesting NRC review of the methodology used in the Wolf Creek categorization as an acceptable methodology for implementing 50.69
  - The PWROG is not seeking approval of changes to the ASME N-660 Code Case that were used at Wolf Creek

Westinghouse

1. Section I-3.0, External Events

Provide description of how piping segments supporting a safe shutdown pathway should be identified and classified.

- Original RI-ISI process considers external events
- Methodology is consistent with NEI 00-04
- WCAP-16308-NP will be revised to correctly reflect the methodology used at Wolf Creek – follows NEI 00-04 process for non-PRA methods and IDP determination process
- No change from endorsed version of Code Case N-660
- Table A-2 will be revised to reflect that Wolf Creek methodology does not differ from endorsed version of N-660

Westinghouse

2. I-3.1.1(a), Break Size Criteria

Identify the criteria that would be used to conclude that the insight does not support a large break

- Proposed change to methodology based on insights from NUREG-1829 and NUREG/CR-4450
- The appropriate break size for consideration in passive categorization is the calculated leak rate at design basis operating conditions for a through-wall flaw with a length 6 times its depth
- Table A-2 will be revised to reflect that Wolf Creek methodology does differs from endorsed version of N-660

Westinghouse

3. I-3.1.2(b), System Impact Group Assessment

a) Identify RI-ISI program criteria referred to in explanatory note

b) describe changes and provide examples illustrating the differences

- Proposed change was not intended to change methodology – Wolf Creek methodology was consistent with endorsed version of N-660
- Table A-2 will be revised to reflect that Wolf Creek methodology does not differ from endorsed version of N-660

Westinghouse

4. General, Interfacing System Support

a) Classifying segments that support multiple functions

b) Handling differences between active and passive categorization results

- Guidance will be added to WCAP ensuring a segment be ranked based on its highest supporting function
- Segments supporting other systems will remain high until the importance of those other systems can be evaluated
- Active and passive rankings should be consistent, however, in certain cases, differences may exist (see examples provided in proposed responses)

Westinghouse

5. Section I-3.2.2(b)(1), HSS Function Consideration

Justify not assigning a HSS classification to an SSC whose failure could consequently fail an HSS function

- Proposed change was not intended to change the methodology – Wolf Creek process was consistent with endorsed version of N-660
- Table A-2 will be revised to reflect that Wolf Creek methodology does not differ from endorsed version of N-660

Westinghouse

6. Section I-3.2.2(b)(2), EOP Considerations

Explain difference between "individually support" and "no redundancy" and define "alternative means of support"

- EOP considerations were consistent with those provided in NEI 00-04
- To ensure consistency between the considerations provided in NEI 00-04 and the passive consideration, Table A-2 will be revised to include questions #4 and #5 from Section 9.2.2 of NEI 00-04

Westinghouse

7. Section I-3.2.2(b), Condition Monitoring Considerations

Provide additional description about how active degradation mechanisms should be incorporated into this process.

- It is proposed that this consideration be removed from the categorization process
- It is not relevant whether or not an active degradation mechanism is present because the consequence assessment is based on a failure probability of 1.0

Westinghouse

8. General, Table A-2 from WCAP-16308-NP

Provide a complete Table A-2 showing all differences between the endorsed version of N-660 and the process used at Wolf Creek

- Supplemental Table A-2 will be revised to be consistent with the methodology used by Wolf Creek
- Additionally, any changes identified to the responses to these RAIs will also be incorporated into Table A-2 and incorporated into a revision of WCAP-16308-NP

Westinghouse

9. Safety Margins

Explain how 10 CFR 50.69(c)(1)(iv) is satisfied and provide reasonable confidence that sufficient safety margins are maintained

- Similar to the active categorization, safety margins are not impacted because there are no proposed changes to the plant design basis
- The categorization process, which includes a consequence assessment and qualitative considerations, assures that changes in CDF and LERF are small
- Also, a licensee is required to show that there is reasonable confidence that any SSC categorized as LSS will be able to reliably perform its design basis function(s) under accident conditions thereby providing an additional level of confidence that the delta CDF and LERF are small

Westinghouse

10. Monitoring of RISC-3 SSCs

Discuss plans to monitor corrective action for degradation of RISC-3 SSCs

- Wolf Creek has not developed plant specific methods for monitoring of correction actions to address degradation of RISC-3 SSCs
- When the plant specific methods are developed, they will follow the approach of NEI 00-04 (see proposed responses for detail) which has been endorsed by NRC and meets the requirements of the 50.69 rule

Westinghouse

11. Application of RISC-3 Treatment Requirements

Discuss plans for inspection, testing, and corrective actions for RISC-3 SSCs

- Wolf Creek has not developed plant specific methods for inspection testing and corrective actions for RISC-3 SSCs
- Wolf Creek will have a formal process for inspection, testing and corrective actions that meets the requirements of 10 CFR 50.69(c)(1)(iv), (d)(2), and (e) consistent with the Commissioner's comments in VR-SECY-04-0109 (see proposed responses for detail)

Westinghouse

Next Steps

The present RAIs are from the review of the passive categorization

- What is the schedule for NRC review of the active categorization?

What are the other steps leading to a draft SE?

Westinghouse