WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard A. Muench
Vice President Technical Services

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ET 01-0028

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

References:
1. Letter ET 01-0009, dated February 15, 2001, from R. A. Muench, WCNOC, to USNRC
2. NRC letter to O. L. Maynard, WCNOC, from J. N. Donohew, USNRC, dated September 5, 2001

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding Relief Request for Application of an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds (TAC No. MB1206)

Gentlemen:

Reference 1 submitted a request for relief from the American Society of Mechanical Engineers (ASME) Section XI code examination requirements for inservice inspection of Class 1 and 2 piping welds. The proposed alternative of a risk-informed inservice inspection (RI-ISI) program is to provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i). Reference 2 provided a request for additional information for the staff to complete its review of the request for relief. Attachment I to this letter provides the requested information. Attachment II contains regulatory commitments made in this submittal.

If you have any questions concerning this matter, please contact me at (620) 364-4034, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

Richard A. Muench

RAM/rlr

Attachments

cc: J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector, w/a

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Provided below is the Wolf Creek Nuclear Operating Corporation (WCNOC) responses to the Nuclear Regulatory Commission questions on the risk-informed inservice inspection (RI-ISI) dated February 15, 2001, for relief from Section XI examination requirements of the American Society of Mechanical Engineers (ASME) Code for Class 1 and 2 piping welds at the Wolf Creek Generating Station (WCGS).

QUESTION 1:

Will the RI-ISI program be updated every 10 years and submitted to the NRC consistent with the current ASME XI requirements?

RESPONSE:

The ISI program will be updated and submitted to the NRC consistent with regulatory requirements in effect at the time such update is required (currently every 10 years). This may again take the form of a relief request to implement an updated RI-ISI program depending on future regulatory requirements.

QUESTION 2:

Under what conditions will the RI-ISI program be resubmitted to the NRC before the end of any 10-year interval?

RESPONSE:

The RI-ISI program will be resubmitted to the NRC prior to the end of any 10-year interval if there is some deviation from the RI-ISI methodology described in the initial submittal or if industry experience determines that there is a need for significant revision to the program as described in the original submittal for that interval. WCNOC will initiate tracking documents to ensure that the RI-ISI program is monitored and periodically reviewed for risk ranking in accordance with the commitments made in Section 4 of the initial submittal. Revisions made as a result of these reviews will be considered for submittal as outlined above.

QUESTION 3:

Page 8 of your submittal presents the criteria for engineering evaluation and additional examinations if unacceptable flaws or relevant conditions are found during examinations. The submittal states that the evaluation will include whether other elements in the segment or segments are subject to the same root cause conditions. The submittal further states that additional examinations will be performed on these elements up to a number equivalent to the number of elements required to be inspected on the segment or segments initially. Please address the following:

a. Please clarify the term "initially". Specifically, does it refer to inspections planned for the current outage or the current interval?

b. Please clarify how will the elements be selected for additional examinations. Specifically, please verify that the elements will be selected based on the root cause or damage mechanism and include high risk significant as well as medium risk significant elements (if needed) to reach the required number of additional elements.

RESPONSE:

a. In this application, the term "initially" refers to those examinations originally scheduled for the current refueling outage.

b. Elements selected for additional examinations will be selected based on the root cause or damage mechanism and will include high risk significant as well as medium risk significant elements (if needed) to reach the required number of additional elements.

QUESTION 4:

Page 4 of your submittal states that a deviation to EPRI RI-ISI methodology has been implemented in the failure potential assessment for thermal stratification, cycling and striping (TASCS). Please state if your revised methodology for assessing TASCS potential is in conformance with the updated criteria described in EPRI letter to NRC dated March 28, 2001. Also, please confirm that as stated in the subject letter, once the final MRP guidance has been developed, the RI-ISI program will be updated for the evaluation of susceptibility to TASCS, as appropriate.

RESPONSE

The methodology for assessing TASCS potential used in the WCGS RI-ISI submittal is identical to the methodology described in Electric Power Research Institute (EPRI) letter to NRC dated March 28, 2001. WCNOC will update the RI-ISI program based on the final EPRI material reliability program guidance as warranted.

QUESTION 5:

Page 13 of your submittal states that WCGS is in the second period of the second interval. The submittal further states that 33% of the ASME XI examinations have been completed thus far, and therefore 67% of the RI-ISI examinations will be performed during the remaining interval so that 100% of the selected examinations are performed during the course of the interval. Please specify which 67% of the RI-ISI examinations will be performed and what will be the basis of the selection.

RESPONSE:

WCGS is currently in the middle of the second period of its second inspection interval. At this point, 33% of the existing ISI program examinations have been completed and 67% of the RI-ISI examinations will be performed during the remainder of the second interval. The examination locations selected by RI-ISI were predicated on contribution to risk and partitioned to appropriately address the various risk categories. The more risk significant welds will be selected for examination within the remainder of this interval.

QUESTION 6:

The NRC safety evaluation (SE) on the WCGS Individual Plant Evaluation (IPE) states that, in their revised IPE human reliability analysis (HRA) submitted to the staff on May 30, 1996, the licensee identified and performed a HRA for a set of five miscalibration actions. However, the staff also noted that the licensee did not provide a basis as to why these were the only five events identified for analysis. What is the basis as to why more miscalibration events were not modeled? Were more miscalibration events modeled in updates to the WCGS probabilistic risk assessment (PRA)?

RESPONSE:

The miscalibration events included in the revised IPE HRA were selected based on a review of events modeled by plants similar to WCGS. Subsequent actions include the addition of miscalibration events for the pressurizer pressure transmitters (BB PT-0455, -0456, -0457, -0458), steam generator level instruments (AE LT-0517, -0518, -0519, -0527, -0528, -0529, -0537, -0538, -0539, -0547, -0548, -0549, -0551, -0552, -0553, -0554) and engineered safety features bus undervoltage instrumentation (NB0101, NB0113, NB0117, NB0116, NB0201, NB0210, NB0216, NB0217) associated with automatic reactor trip, safety injection, auxiliary feedwater actuation, and loss of offsite power signals. Addition of these miscalibration events had negligible impact on core damage frequency (CDF) results.

A review of modeled human actions was also performed to determine the impact of instrument miscalibration failures on the calculated HRA event values. The review concentrated on operator actions based solely on the reading of a single instrument or single group of instruments. Based on sensitivity evaluations performed, no HRA events were identified where instrument miscalibration failures would have a significant CDF impact (maximum of 0.15 percent CDF increase). The review identified that for many of the operator actions, more than one cue is available.

A review of WCGS Licensee Event Reports (LERs) for the last ten years revealed no LERs associated with the miscalibration of instruments which can have a significant impact on CDF.

QUESTION 7:

The NRC SE on the Wolf Creek IPE further states that the licensee's revised HRA analysis does not identify and analyze time-critical actions (actions where the difference between the time available and the time required to perform the actions is short and the possibility exists for the operators to fail to accomplish the actions in time is significant). The staff also states that the licensee provides some information concerning the time available for performing actions, but the licensee does not provide information concerning the time required to accomplish the actions. Have more recent updates in the Wolf Creek PRA improved the modeling of time in recovery actions that must be performed within a very short time?

RESPONSE:

Time critical actions are defined as those which take a long time to diagnose and perform, relative to the length of the time window available. The time critical actions are currently treated, and were treated for the last PSA mode update in the following manner in the HRA:

Identification: Time critical actions are primarily identified through the operator interview process, and an examination of the time windows available from thermal-hydraulic analyses such as MAAP or other engineering calculations. The operator interview process ascertains the cues and steps in the procedure that the operators use to diagnose the event and the time at which this diagnosis takes. Then, the steps judged to be critical to that particular HRA are confirmed and the overall time to successfully complete these steps determined. The overall time accounts for potential delays due to additional, non-critical procedural steps that must be executed first, time required for the component to change state (e.g., to start a turbine-driven pump), and limitations that may be present due to operator crew manning.

Treatment: If the time window is less than the diagnosis time plus the time required to successfully complete the actions, then the action is assumed to be failed. If the time window is larger than the diagnosis time plus the time required to successfully complete the actions, then the probability of failure is adjusted either directly (e.g., taken as an unavailability of 0.1), or through selection of the stress factor and the allowed credit for recovery. For example, if there is a 30 minute time window and the action takes 5 minutes to diagnose and 15-20 minutes to execute, then a moderate to extreme level of stress is taken (depending on if there are other, competing actions occurring simultaneously) and no credit is given for recovery. Alternately, if the time window is 1 hour, and the action is at the end of a success branch on an event tree (e.g., LOCA followed by successful injection, cooldown, and depressurization such that the time window starts several hours after the initiator), and the competition from other actions is low, then the stress is taken as optimal and credit may be given for recovery. In each case the operator actions are examined in the full context of the scenario, including timing, to determine the potential impact of time constraints.

QUESTION 8:

The NRC SE on the Wolf Creek IPE also states that the licensee identified five modifications in their IPE that would decrease core damage frequency (CDF), if implemented. Two of these items were credited in the IPE, although they had not yet been implemented. Have all of the five modifications been implemented and does the current PRA reflect the implementation of these modifications?

RESPONSE:

The implementation status for each of the five modifications identified in the NRC SE on the WCGS IPE is addressed in items a. through e. below. The status of each of these modifications relative to the current probabilistic safety analysis (PSA) is also indicated.

a. WCGS currently has high temperature qualified seal materials installed for three of four Reactor Coolant Pumps (RCP). The current schedule calls for installation of seals containing high temperature qualified materials for the remaining RCP in Refuel 13 (Fall 2003). The RCP seal loss-of-coolant accident (LOCA) model utilized for the current WCGS PSA model is a Westinghouse RCP Seal LOCA Model developed during the IPE using information from WCAP-10541, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," along with consideration for the hypothetical catastrophic "seal ring binding" and "seal popping" failure mechanisms. The current WCGS PSA model applies parameters determined from the Westinghouse RCP Seal LOCA Model for the IPE, which reflects a configuration where all four RCP seals contain the older unqualified materials. Simple sensitivity evaluations of the current core damage results were performed using information from Brookhaven National Laboratory (BNL) Technical Report W6211-08/99, "Guidance Document for Modeling of RCP Seal Failures" (Brookhaven), and Appendix A of NUREG/CR-5167, "Cost/Benefit Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure" (Rhodes). For the current configuration of three seals with qualified materials and one seal with unqualified materials, a CDF reduction of approximately 12% was estimated using Brookhaven information and a CDF reduction of approximately 6% estimated using Rhodes information. The Westinghouse Owners Group (WOG) submitted WCAP-15603, "WOG2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," on December 20, 2000, for NRC for review and approval. WCNOC currently plans to apply this RCP Seal Leakage Model, once approved by the NRC, in the WCGS PSA model.

b. The positive displacement charging pump has been replaced with a centrifugal charging pump which is identified as the Normal Charging Pump (NCP). The electric motor driver for the NCP is supplied from a non-safety related 4160 Volt AC load center. The NCP does not have a direct operational dependency on an external cooling water source. The current WCGS PRA model does not reflect installation of the NCP. WCNOC currently plans to incorporate the operation of the NCP into an ongoing PSA model update.

c. A modification to allow for bypass of a feedwater isolation signal in order to restore main feedwater has been implemented at WCGS. As indicated in the NRC SE for the WCGS IPE, credit for this modification has been included in the WCGS PRA model since the time of the IPE.

d. Procedures are in place to address loss of component cooling water and loss of service water conditions. These procedures provide direction for alignment of an alternate cooling water source, using staged equipment, for lube oil cooling for the centrifugal charging and safety injection pumps. The current WCGS PSA model reflects actions to align an alternate cooling water source (fire protection water) for these pumps as directed by procedures OFN EG-004, "CCW System Malfunctions," or OFN EF-033, "Loss of Essential Service Water."

e. Procedural guidance is provided in Attachment C to EMG C-0, "Loss of All AC Power," to shed selected DC loads during a Station Blackout event to extend battery life. As indicated in the NRC SE for the WCGS IPE, credit for this procedural action has been included in the WCGS PRA model since the time of the IPE.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4038.

COMMITMENT	Due Date/Event
The RI-ISI program will be resubmitted to the NRC prior to the end of any 10-year interval if there is some deviation from the RI-ISI methodology described in the initial submittal or if industry experience determines that there is a need for significant revision to the program as described in the original submittal for that interval. WCNOC will initiate tracking documents to ensure that the RI-ISI program is monitored and periodically reviewed for risk ranking in accordance with the commitments made in Section 4 of the initial submittal. Revisions made as a result of these reviews will be considered for submittal as outlined above.	Concurrent with the implementation of the approved relief request.
The methodology for assessing TASCS potential used in the WCGS RI-ISI submittal is identical to the methodology described in Electric Power Research Institute (EPRI) letter to NRC dated March 28, 2001. WCNOC will update the RI-ISI program based on the final EPRI material reliability program guidance as warranted.	Upon issuance and review of the final EPRI material reliability program.
At this point, 33% of the existing ISI program examinations have been completed and 67% of the RI-ISI examinations will be performed during the remainder of the second interval. The examination locations selected by RI-ISI were predicated on contribution to risk and partitioned to appropriately address the various risk categories. The more risk significant welds will be selected for examination within the remainder of this interval.	Concurrent with the implementation of the approved relief request.