WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J Garrett
Vice President, Engineering

July 12, 2006

ET 06-0027

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

Reference: Letter ET 06-0011, dated March 2, 2006, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket 50-482: Wolf Creek Nuclear Operating Corporation's Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

Gentlemen:

The Reference provided Wolf Creek Nuclear Operating Corporation (WCNOC) 10 CFR 50.55a Requests I2R-34, I2R-35, I2R-36, I2R-37, and I2R-38, which requested alternatives to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI for inservice inspection and testing for the Second Ten-Year Interval of WCNOC's Inservice Inspection (ISI) Program.

On May 9, 2006, the Nuclear Regulatory Commission (NRC) Project Manager for WCNOC provided by electronic mail a request for additional information (RAI) regarding 10 CFR 50.55a (Relief) Requests I2R-34, I2R-35, and I2R-36. On May 31, 2006, the questions in that RAI were discussed in a telephone call between the NRC Project Manager for WCNOC, the NRC reviewer of the subject requests, and WCNOC staff.

The Attachment to this letter provides WCNOC's response to the RAI. It lists each NRC question followed by WCNOC's response to each of those questions.

A047

There are no commitments associated with this submittal. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Very truly yours,

Terry J. Garrett

TJG/rlt

Attachment

cc: J. N. Donohew (NRC), w/a
W. B. Jones (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

1. Request for Additional Information:

Provide the following information related to the welds that are addressed in the RR I2R-34 and RR I2R-35.

A. Previous ultrasonic testing (UT) results which include pre-service and prior in-service examination of each weld.

Response:

The welds listed in the subject relief requests (RRs) were examined Pre-service and in Interval 1. The examination results are listed in the below table.

Weld	Preservice	Interval 1
TBB03-10A-W	NI	NI
TBB03-10C-W	NRI	NRI
TBB03-10C-IR	NI	NI
RV-101-141	4 RI*; all acceptable; IWB-3510-1	NI *
RV-102-151	NRI	NRI

NI= No indication
NRI= No recordable indication (geometry noted)
RI= Recordable indication
***The differences in the preservice and Interval 1 exams is attributable to the advances in examination techniques and equipment in this time.**

B. The extent of the previous UT examination coverage for each weld.

Response:

The extent of examination for each weld as recorded on the data sheets is listed in the table below:

Weld	Preservice	Interval 1
TBB03-10A-W	Limited	75.08%
TBB03-10C-W	Complete	Complete
TBB03-10C-IR	Complete	Complete
RV-101-141	Limited	65.3%
RV-102-151	Limited	48.2%

Note: The coverage listed is a composite coverage.

Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

C. Previous industry experience associated with aging degradation of the subject welds.

Response:

There have been no reported instances of service induced failures or degradation of the subject welds in the industry. A search of the Wolf Creek database that tracks industry issues was performed. No operating experience (OE) was found on failures of the subject welds. A search on the INPO website for OE on the subject welds was also performed. Various ASME Section XI code committee actions were also researched. All data indicates that no service induced failures or degradation of the subject welds have occurred in the industry.

There is no degradation mechanism, other than fatigue, that is associated with these welds or areas. The subject welds are not subject to Stress Corrosion Cracking (SCC).

D. Type of weld metal (i.e., carbon/low alloy steel, nickel alloy or stainless steel) that was used for each weld.

Response:

All of the subject welds are carbon/low alloy steel. Note that TBB03-10C-IR is not a weld, it is an inner radius area. A stainless steel cladding has been applied to the inside surface of all of the subject areas.

2. Request for Additional Information:

For RRs I2R-34, I2R-35, and I2R-36, provide information regarding the state of stress that is present in the volume of each weld that will not be included in the future examinations. Clarify whether the highly stressed volume of the weld, where cracks can initiate, can be examined effectively with the proposed alternative. If this is not the case, provide a justification for not including the highly stressed volume in the examination. Provide information regarding the aging monitoring program (e.g., plant's leakage monitoring program) which can adequately identify any service-induced cracking that can originate from the highly stressed uninspected regions of the welds, and provide assurance that proper corrective actions for restoring the structural integrity of the welds will be implemented if such cracking is identified.

Response:

The above has been broken out and each part answered individually.

"...provide information regarding the state of stress that is present in the volume of each weld that will not be included in the future examinations."

Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

Wolf Creek Nuclear Operating corporation (WCNOC) has not performed finite element analysis on the components listed in the subject RRs, therefore a stress profile across the welds is not available. Per the stress reports and the spec sheets, the stresses of the subject welds are all within the Code allowables. Please note that these RRs are not for future examinations; they reflect the Interval 2 examinations which were performed to the maximum extent possible based on specified requirements. Future examinations in Interval 3 will be based on specified requirements applicable at those times, and if needed, relief will be requested at that time.

"Clarify whether the highly stressed volume of the weld, where cracks can initiate, can be examined effectively with the proposed alternative. If this is not the case, provide a justification for not including the highly stressed volume in the examination."

For the components addressed in I2R-34:

TBB03-10C-W, pressurizer spray nozzle to vessel weld: The location for which the examination is limited is the adjacent base metal on the nozzle side of the weld. This nozzle base metal is a lower stress region of the examination volume typical for this configuration. The entire volume of the weld, where degradation would be first expected to occur, was examined to the Code requirements.

TBB03-10A-W, the pressurizer surge nozzle to vessel weld: The location for which the examination is limited is the adjacent base metal and a very small part of the weld itself. The limitation in weld was limited only for the 60 degree perpendicular scan, however, this volume was examined with the 45 degree perpendicular scan, as shown in the attached Figure 1 to I2R-34. The base metal limitations were shown in Figure 1 and Illustrate that much of the base metal (which would not be subject to a degradation mechanism) has been examined. Figure 1 also illustrates that the volume of the weld where degradation would first be expected to begin has been adequately examined.

The proposed alternative is: 1) UT of the subject weld was performed to the maximum extent, 2) VT-2 visual examinations per Code Category B-P (this examination is performed every outage). If cracking of these components were to initiate, it is expected to initiate at the inside diameter (ID). Any significant cracking would have likely been detected by the performed examination. In addition, any through wall leakage of these components during the cycle would be detected by the VT-2 pressure test and, due to the high toughness of these welds, would be expected to be identified well before critical flaw size is reached. Therefore, the proposed alternative provides reasonable assurance of structural integrity.

For the components addressed in I2R-35:

The stresses are relatively low in the subject welds. As stated above, the only degradation mechanism that might affect these areas is fatigue. Per the stress report, the maximum usage factor of these areas is 0.007, which compares very favorably with 1.0.

Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

The proposed alternative is: 1) UT of the subject weld performed to the maximum extent possible (over 70% composite coverage of the code required volume (CRV) was achieved), 2) Visual examination performed as required per Code Category B-N-1 (the VT-3 visual examination was performed on the ID of the subject welds in the last outage), and 3) VT-2 visual examinations per Code Category B-P (this examination is performed every outage). As stated above, the maximum usage factor for these welds is extremely low and when combined with the stresses being relatively low, the basis is formed that the proposed alternative provides reasonable assurance of structural integrity.

For the component addressed in I2R-36:

A stress report was not required as this is a Class 2 component; however the spec sheet states that the design specification (which includes allowable stresses) has been met. In addition, an Industry paper (reference 1) was reviewed. A review of this paper revealed the following: A finite element model was developed for a generic residual heat removal (RHR) heat exchanger and stress calculations were carried out as a basis for flaw tolerance calculations. Fracture calculations were carried out to determine the critical flaw depth for the welds that are examined. In all cases, the critical flaw depth exceeded the wall thickness. The fracture calculations were extended to consider the critical length for a through wall flaw. In all cases, the critical length was found to exceed the diameter of the heat exchanger. The conclusion of the paper is that the fracture evaluation shows there is a very large flaw tolerance in these heat exchangers, and that they experience a very mild duty cycle.

The proposed alternative is: 1) UT of the subject weld performed to the maximum extent possible and 2) VT-2 visual examinations per Code Category C-H. The entire (100%) CRV was examined by perpendicular angle beam scan (scanning for circumferential flaws) and 50% of the CRV examined by parallel angle beam scan (scanning for axial flaws). With the large flaw tolerance in the RHR heat exchanger, the UT examinations performed would detect flaws well before their size presented any safety concern. The RHR heat exchanger is not insulated and leakage through this component would be readily detected by the VT-2 pressure test. Therefore, the proposed alternative provides reasonable assurance of structural integrity.

"Provide information regarding the aging monitoring program (e.g., plant's leakage monitoring program) which can adequately identify any service-induced cracking that can originate from the highly stressed uninspected regions of the welds."

There are no regions uninspected that are judged to be highly stressed. The applicable aging monitoring program to identify service induced cracking is ASME Section XI, which requires UT examinations and VT-2 visual examinations for evidence of through wall leaks. The Code UT examinations have been performed to the greatest extent practical and the Code visual examinations are performed as required. Therefore, the proposed alternative provides reasonable assurance of structural integrity.

Response to Request for Additional Information Regarding 10 CFR 50.55a Requests I2R-34, I2R-35, and I2R-36

"...and provide assurance that proper corrective actions for restoring the structural integrity of the welds will be implemented if such cracking is identified."

The WCNOC Technical Requirements Manual section 3.4.17 and administrative procedures address nonconformances and require corrective actions when structural integrity issues are identified.

3. Request for Additional Information:

The submittal indicates that visual testing (VT-2) examination is performed during a pressure test. According to paragraph IWA-5242 of the ASME Code Section XI, during a pressure test of a system with borated water, insulation shall be removed so that any reactor coolant system (RCS) leakage can be detected during the VT-2 examination. This requirement applies to Category B-D welds (RR I2R-34) and Category B-A welds (RR I2R-35). Confirm whether the insulation is removed during pressure tests and, if it not, explain how any RCS leakage is effectively identified during the pressure tests.

Response:

This question was withdrawn during a phone call on 5/31/2006 between the NRC Project Manager for WCNOC, the NRC reviewer of the subject requests, and WCNOC staff.

Reference:

1. Technical Basis for Revision of Inspection Requirements for Regenerative and Residual Heat Exchangers; August, 2004; *Document required under WOG project MHUP 5093*; ASME BPVC action, ISI-03-06, BC03-338