Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

March 4, 2011
LIC-11-0016

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

References:
1. Docket No. 50-285
2. Letter from OPPD (H. J. Faulhaber) to NRC (Document Control Desk), "10 CFR 50.55a Request Number RR-12, Omaha Public Power District (OPPD) Request for Relief from Code Case N-722 Visual Examination (VE) of the Reactor Vessel Hot Leg Nozzle to Safe End Dissimilar Metal Welds," dated August 16, 2010 (LIC-10-0065) (ML102300641)
3. Letter from NRC (L. Wilkins) to OPPD (D. J. Bannister), "Fort Calhoun Station, Unit No. 1 – Request for Additional Information Re: Request for Relief From Code Case N-722 Visual Examination of the Reactor Vessel Hot-Leg Nozzle-to-Safe End Dissimilar Metal Welds (TAC No. ME4541)," dated October 25, 2010 (NRC-10-0092)
4. Letter from OPPD (H. J. Faulhaber) to NRC (Document Control Desk), "Response to Request for Additional Information (RAI) Regarding Request for Relief from ASME Code Case N-722 Requirements for Visual Examination of Reactor Vessel Hot Leg Nozzle to Safe End Welds," dated November 4, 2010 (LIC-10-0095)
5. Email from NRC (L. Wilkins) to OPPD (B. Hansher), "DRAFT 2nd RAI for Fort Calhoun Station, Unit 1 Relief Request (TAC No. ME4541)," dated February 23, 2011 (NRC-11-0013) (ML110540441, ML110540449)

SUBJECT: Response to Second NRC Request for Additional Information (RAI) Regarding Request for Relief from ASME Code Case N-722 Requirements for Visual Examination of Reactor Vessel Hot Leg Nozzle-to-Safe End Dissimilar Metal Butt Welds

In support of the Omaha Public Power District's (OPPD) Reference 2 Relief Request, attached is OPPD's response to the NRC's second request for additional information (Reference 5).

No commitments to the NRC are contained in this submittal.

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NRR

If you have any questions regarding this submittal, please contact Mr. Bill Hansher at (402) 533-6894.

Sincerely,

John B. Herman
Division Manager-Nuclear Engineering

JBH/BL/mle

Attachment: OPPD Response to NRC Second Request for Additional Information

c: E. E. Collins, NRC Regional Administrator, Region IV
L. E. Wilkins, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector

OPPD Response to NRC Second Request for Additional Information

SECOND REQUEST FOR ADDITIONAL INFORMATION

By letter dated August 16, 2010 as supplemented by letter dated January 14, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML102300641 and ML110200193), Omaha Public Power District (the licensee), proposed an alternative to 10 CFR 50.55a(g)(6)(ii)(E) for Fort Calhoun Station, Unit 1. This requirement defines the inservice inspection frequency of visual examination of the reactor vessel hot leg nozzle to safe end welds in accordance with American Society of Mechanical Engineer's Boiler and Pressure Vessel Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials," with NRC conditions. The duration of request is for the fourth 10-year ISI interval, which ends on September 25, 2013.

The NRC staff has reviewed and evaluated the information provided by the licensee and has determined that the following information is needed in order to complete its review of the relief request.

NRC Question

1. In Reference 8.a of the Attachment to the August 16, 2010 letter, the as built dimension of the Nozzle Extension Forging (a.k.a. safe end) is not provided. The drawing identifies a distance much larger than assumed in the analysis under Ref 9, 10 and 11.

   Provide the basis for the distance between the dissimilar metal weld and stainless steel weld used in the analysis.

OPPD Response

The dimension of the final machined nozzle extension forging shown in Reference 8.a of the Attachment to the August 16, 2010 letter is approximately 1/8" longer than the as-built final machined dimension. The distance between the dissimilar metal weld and stainless steel weld is longer than was used in the Reference 9, 10 and 11 analyses.

The inherent conservatism of the residual stress profiles associated with a 25% inside surface weld repair is sufficient to offset the potential reduction in stress relieving effects associated with a longer distance between the dissimilar metal weld and stainless steel weld. Based on a search of the weld repair fabrication records and empirical eddy current pancake coil data, it is concluded that no weld repairs were made to either of the Fort Calhoun Station (FCS) reactor vessel (RV) outlet nozzle dissimilar metal butt welds (DMBWs). Therefore, the assumption of a

25% initial inside surface weld repair in the residual stress analysis provides an inherent conservatism in the residual stress profiles shown in Figure 4-1 of the submittal. The significant level of conservatism associated with the assumption of a 25% inside surface weld repair can be seen in the hoop and axial stress plots shown respectively in Figures 21 and 22 of Reference 9. Figures 21 and 22 provide the through-wall residual stress distributions for various steps of the weld fabrication process, namely:

- after the completion of the initial DMBW,
- after the assumed 25% inside surface weld repair, and
- after post weld heat treatment (PWHT).

Figures 21 and 22 show that after the completion of the initial DMBW, the residual stress at the inner region of the weld is significantly lower than after the assumed inside surface weld repair.

PWHT performed immediately after the completion of the initial DMBW (without considering any inside surface weld repair), is similar to the presence of a very long safe end without the stress relieving effects from an adjacent stainless steel safe end weld. The resulting residual stress is expected to be much lower than that shown in Figures 21 and 22 for a 25% inside surface weld repair with PWHT.

In conclusion, the inherent conservatism of the residual stress profiles shown in Figure 4-1 of the submittal is sufficient to offset the potential reduction in stress relieving effects associated with a longer distance between the dissimilar metal weld and stainless steel weld.

NRC Question

2. It appears from Ref 6 of the Attachment to the August 16, 2010 letter, that the post EPU loads used in the analysis as documented in Table 3-3 were not conservative. Loop 2 loads appear to be used for Deadweight + Normal Operating Thermal, and Loop 1 values were used for Operational Basis Earthquake and Safe Shutdown Earthquake. However, initial review of Reference 6 indicates the values calculated for the other Loop in each case were higher.

   State the basis for using minimum Loop loading values in Table 3-3 of the Attachment to the August 16, 2010 letter.

OPPD Response

The deadweight + normal operating thermal loads (lbs, in-lbs) were obtained from page 38 of Reference 6.b of the Attachment to the August 16, 2010 letter. The resultant moment was determined for Loop 1 and 2 as shown in the table below since resultant moment, not individual moment components, is used in the evaluation. Loop 2 has the higher resultant moment by a negligible amount. The impact of the slightly lower Loop 2 F_x load is insignificant compared to the high axial force due to normal operating pressure. Therefore, as indicated by the bold font in the table below, Loop 2 loads (kips, in-kips) were used and tabulated in Table 3-3.

	F_x (lbs)	M_x (in-lbs)	M_y (in-lbs)	M_z (in-lbs)	Resultant Moment (in-lbs)
Loop 1	4504.98	-10553.11	20367.89	-23922993.16	23923004
Loop 2	**4240.09**	**-7398.04**	**-2679.71**	**23930410.49**	**23930412**

The operating basis earthquake (OBE) and safe shutdown earthquake (SSE) loads (kips, ft-kips) were obtained from page 60 of Reference 6.c of the Attachment to the August 16, 2010 letter. The resultant moment was determined for Loop 1 and 2 as shown in the table below since resultant moment, not individual moment components, is used in the evaluation. Loop 1 has the higher resultant moment by a negligible amount. The impact of the slightly lower Loop 1 F_x load is insignificant compared to the high axial force due to normal operating pressure. Therefore, as indicated by the bold font in the table below, Loop 1 loads (kips, in-kips) were used and tabulated in Table 3-3.

	F_x (kips)	M_x (ft-kips)	M_y (ft-kips)	M_z (ft-kips)	Resultant Moment (ft-kips)
Loop 1 (OBE)	**108.03**	**96.08**	**179.58**	**178.74**	**270.98**
Loop 2 (OBE)	109.40	91.92	181.25	177.76	270.00

	F_x (kips)	M_x (ft-kips)	M_y (ft-kips)	M_z (ft-kips)	Resultant Moment (ft-kips)
Loop 1 (SSE)	**161.05**	**154.75**	**266.91**	**257.44**	**401.83**
Loop 2 (SSE)	162.61	155.70	268.93	253.74	401.19

It should be noted that the individual moment components and the resultant moment (table column labeled as SRSS [square root sum of the squares]), summarized on page 60 of Reference 6.c of the Attachment to the August 16, 2010 letter, represent the maximum of all the time steps. Therefore, the use of maximum individual moment components tabulated on page 60 of Reference 6.c of the Attachment to the August 16, 2010 letter in determining the resultant moment is conservative.

In conclusion, the minimum loop loading values are not used in the analysis. In fact, the loads for the loop with the higher resultant moment are used.

NRC Question

3. Figures 18 and 19 of Reference 11 of the Attachment to the August 16, 2010 letter appear to be the closest stress profiles to the one used to develop Figure 4-1 of the submittal. Figures 18 and 19 included post weld heat treatment, a 25% weld repair and a 5-13/32 inches safe end length.

   Provide the basis for the differences between the analysis and results between Figures 18, 19 and 4-1. Specifically, state the basis for the reduction in magnitude of the hoop and axial stresses in the inner half thickness of the nozzle.

OPPD Response

Figures 18 and 19 of Reference 11 are not the residual stress profiles used to develop Figure 4-1. Figure 4-1 of the submittal is based on the hoop and axial stress tabulation along Path 2 on page 18 of Reference 9 of the Attachment to the August 16, 2010 letter. This represents the normal operating condition through-wall residual stress distribution for the DMBW with post weld heat treatment and a 25% inside surface weld repair.

In conclusion, there was no reduction made in the stress magnitudes from Reference 9.

NRC Question

4. Please, describe the post weld heat treatment procedure used after the dissimilar metal weld was fabricated.

OPPD Response

The PWHT temperature for fabrication of the RV nozzle (inlet and outlet) DMBWs was 1150°F, for a period of 1-hour minimum per inch of thickness, which is consistent with the ASME Code requirements. The RV nozzle low-alloy steel and stainless steel safe-end were welded to each other using Alloy 82/182 weld material (i.e., the DMBW). The PWHT performed at FCS after the joining process was completed differs from that used on the RV nozzles at most other U.S. pressurized water reactors (PWRs). For these other RV nozzles, a thin Alloy 82/182 buttering was first welded to the low alloy steel material; PWHT was then performed, and finally, the RV nozzle and safe-end were welded together using Alloy 82/182 weld material without any further PWHT.

At FCS, there is no Alloy 82/182 buttering and the entire Alloy 82/182 DMBW received PWHT. In contrast, only the Alloy 82/182 buttering on the RV nozzles received PWHT at most other U.S. PWRs. PWHT of the entire Alloy 82/182 DMBW at FCS results in lower residual stresses at the DMBW, which reduces the potential for initiation of primary water stress corrosion cracking (PWSCC).

Additionally, a review of fabrication records from the manufacturer (Combustion Engineering) of the RV found no evidence that any weld repairs were made to the FCS RV outlet nozzle DMBWs. The absence of any inside surface weld repairs to the DMBWs on the RV outlet nozzles during manufacturing is further substantiated by the empirical eddy current testing (ECT) pancake coil data taken during the 2009 FCS refueling outage. ECT pancake coil data shows a lack of significant permeability changes in both RV outlet nozzle DMBWs, which confirms that no prior weld repairs were undertaken. Permeability changes were found some distance away from the DMBWs in the stainless steel cladding where minor repairs were made. These observed changes validate the technology as being able to associate permeability changes with weld repairs in the DMBWs.

For the current analysis, PWHT of the DMBW was simulated by raising the temperature of the finite element model to a PWHT temperature of 1100°F. The creep-related stress relaxation effect of time at temperature was not considered in the simulation. This is an industry best practice used in the simulation of the PWHT process. The lower residual stress shown in Figure 4-1 of the submittal are consistent with those associated with PWHT welds. The lower residual stress shown in Figure 4-1 is consistent with ECT inspection results from the 2009 refueling outage at FCS, which show that no surface connected indications exist.