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

LIC-14-0005
January 28, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

References:
1. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), "License Amendment Request 13-06; Plant-Specific Leak-Before-Break Analysis," dated August 5, 2013 (LIC-13-0100) (ML13220A073)
2. Email from NRC (Lynnea Wilkins) to OPPD (B. R. Hansher), "DRAFT: Fort Calhoun RAI Re: Leak Before Break LAR (TAC MF2559)," dated November 8, 2013 (ML13316A054) (NRC-13-0142)

SUBJECT: Response to NRC Request for Additional Information (RAI) Re: Leak Before Break LAR (TAC MF2559)

Attached is the Omaha Public Power District (OPPD) response to the Nuclear Regulatory Commission (NRC) request for additional information (RAI) (Reference 2) regarding the Leak-Before-Break analysis submitted by Reference 1.

There are no new regulatory commitments being made in this letter.

In accordance with 10 CFR 50.91, a copy of this letter, without the attachment, is being provided to the designated State of Nebraska official.

If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher, Supervisor-Nuclear Licensing, at 402-533-6894.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 28, 2014.

Louis P. Cortopassi
Site Vice President and CNO

LPC/KGM/mle

Attachment: OPPD Response to NRC RAI Re: Leak Before Break LAR (TAC MF2559)

c: M. L. Dapas, NRC Regional Administrator, Region IV
J. W. Sebrosky, NRC Senior Project Manager
J. C. Kirkland, NRC Senior Resident Inspector
Manager Radiation Control Program, Nebraska Health & Human Services, R & L Public Health Assurance, State of Nebraska (w/out attachments)

Omaha Public Power District (OPPD) Response to NRC Request for Additional Information (RAI) Re: Leak Before Break LAR (TAC MF2559)

By letter dated August 5, 2013, Omaha Public Power District (the licensee) submitted for Nuclear Regulatory Commission (NRC) review and approval a license amendment request relating to the plant-specific leak-before-break (LBB) analysis for the reactor coolant system (RCS) primary loop piping at Fort Calhoun Station, Unit 1. The licensee submitted the plant-specific LBB analysis to satisfy one of its commitments as part of its license renewal application before the period of extended operation, which began at midnight on August 9, 2013. The licensee's LBB analysis is documented in Westinghouse report, WCAP-17262-P, Revision 1. The NRC staff notes that it approved the RCS primary loop piping at Ft Calhoun for LBB under a generic application on February 1, 1984.

The NRC staff has reviewed your submittal and has determined that the information specified in the Request for Additional Information (RAI) below is needed for the staff to complete its evaluation.

Please contact me if a clarifying teleconference is needed for the attached RAIs.

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST 13-06

LEAK BEFORE BREAK APPLICATION FOR

REACTOR COOLANT PRIMARY LOOP PIPING

FORT CALHOUN STATION UNIT 1

OMAHA PUBLIC POWER DISTRICT

DOCKET NUMBER 50-285

NRC RAI 1

The Westinghouse report, WCAP-17262-P, identified several nickel-based Alloy 82/182 welds in the RCS primary loop piping. The industry operating experience has shown that primary stress corrosion cracking (PWSCC) is an active degradation mechanism in Alloy 82/182 weld material. Standard Review Plan 3.6.3 specifies that LBB cannot be applied to piping that experiences an active degradation mechanism. On page 5 of the Enclosure to the August 5, 2013 letter, the licensee stated that mitigation will be implemented to minimize PWSCC at the reactor vessel nozzle locations which have nickel-based Alloy 82/182 welds.

1. Please discuss the type of mitigation that will be implemented and when the mitigation will be implemented.

   OPPD Response

   At Fort Calhoun Station, access is very limited to the outside diameter of the six reactor vessel nozzles, consequently any outside diameter mitigation such as weld overlay or mechanical stress improvement is not feasible. Mitigation will therefore be implemented from the inside diameter surface using either underwater laser beam welding or by applying a compressive stress to the inside diameter surface by using water jet peening or laser ablation. Mitigation will be implemented during the fall 2019 refueling outage.

2. Please provide the weld number for each of the Alloy 82/182 welds in the RCS primary loop piping.

OPPD Response

Welds at Locations 1 and 14 (Nozzle to safe end). Figure 3-2 of WCAP-17262-P, Revision 1 identified the weld locations.

3. Please discuss previous inspection history and results of these welds, including whether the ultrasonic examinations of these welds achieve essentially 100-percent coverage of required volume.

OPPD Response

The six reactor vessel nozzle dissimilar metal Alloy 82/182 butt welds were inspected as part of the In-service Inspection (ISI) Program on a ten-year frequency. Additional inspections were added due to the requirements of Electric Power Research Institute (EPRI) Material Reliability Program (MRP)-139, which has now been superseded by Code Case N-770. The last two inspections, (i.e., the ISI 10-year inspection in 2003 and the MRP-139 inspection in 2009) are significant because these inspections met the requirements of the Performance Demonstration Initiative. The 2009 inspections were performed by WesDyne ("Fort Calhoun Reactor Vessel Nozzle Examinations, Outage R025 2009," WesDyne Report Number WDI-PJF-1304080-FSR-001, C.S. Wyffels, November 16, 2009) using SUPREEM (Submersible Platform with ROSA End Effector Motion) robotic technology. The 2009 internal diameter examinations used a combination of ultrasonic testing, plus point eddy current testing, and pancake eddy current testing. The pancake eddy current testing showed that the weld locations were in close agreement with as-built engineering drawings. The 2009 examinations achieved 100% coverage, and no recordable indications were found in any of the six reactor vessel nozzle dissimilar metal Alloy 82/182 butt welds.

4. Discuss future inspections of the Alloy 82/182 welds.

OPPD Response

During the spring 2015 refueling outage, the six reactor vessel nozzle dissimilar metal Alloy 82/182 butt welds will be volumetrically inspected using a combination of ultrasonic testing and eddy current testing.

NRC RAI 2

Cracks caused by PWSCC or fatigue may occur in the Alloy 82/182 weld. The PWSCC crack morphology is different from the fatigue crack morphology. Therefore, calculating the leak rate from a PWSCC crack would be different from that of a fatigue crack. It appears that WCAP-17262-P (page 6-1) calculates the leak rate based on fatigue crack morphology. It appears that WCAP-17262-P also considered PWSCC crack morphology in the leak rate calculation as discussed on Page 6-3 of WCAP-17262-P.

Please confirm that PWSCC crack morphology was used in the leakage rate calculation for the Alloy 82/182 weld and fatigue crack morphology was used in the leakage rate calculation for the cast austenitic stainless steel pipe.

OPPD Response

That is correct. Primary water stress corrosion crack (PWSCC) crack morphology was used in the leakage rate calculation for the Alloy 82/182 weld and fatigue crack morphology was used in the leakage rate calculation for the cast austenitic stainless steel pipe.

NRC RAI 3

Section 2.1 of WCAP-17262-P briefly touched upon PWSCC in Alloy 82/182. Please assess and discuss in detail:

1. The impact of PWSCC on the RCS primary loop piping in light of existing Alloy 82/182 welds;

   OPPD Response

   Please see response to 3 below.

2. How PWSCC will be minimized until the Alloy 82/182 welds are mitigated;

   OPPD Response

   For the next five years, no power uprates are planned for Fort Calhoun Station. Thus, T_{hot} will remain at its current value and with no temperature increase from a power uprate, the occurrence of PWSCC is minimized.

   Furthermore, zinc injection was started at Fort Calhoun Station in June 2002 and continues to be performed. Currently, zinc additions average 6 parts per billion (ppb) per month, and a total of 411 ppb-months have been added into the reactor coolant system since program inception in 2002. This value of 411 ppb-months of zinc exceeds the required level of 300 ppb-months, which is known to help mitigate PWSCC. Zinc injections will continue until the six reactor vessel nozzle dissimilar metal Alloy 82/182 butt welds are fully mitigated.

3. How structural integrity of the unmitigated Alloy 82/182 welds will be maintained.

   OPPD Response

   LBB analysis was performed at the Alloy 82/182 weld location considering the effects of PWSCC parameters. LBB margin of 10 on leak rate, margin on flaw size of >2 and margin on loads of 1 (using absolute summation method of faulted load combination) were demonstrated and therefore structural integrity of the unmitigated Alloy 82/182 welds will be maintained.

NRC RAI 4

The licensee calculated two different sets of flaw sizes at Location 1 which is the joint connecting the cast austenitic stainless steel pipe to the carbon steel reactor vessel nozzle by an Alloy 82/182 weld. For the cast austenitic stainless steel pipe, the licensee calculated a leakage crack size and critical crack size of 5.28 inches and 35.27 inches, respectively. For the Alloy 82/182 weld, the licensee calculated a leakage crack size and critical crack size X and Y inches (proprietary information).

1. Please provide detailed drawings of the Location 1 configuration, including the nozzle, weld, safe end, and pipe. The drawing should include dimensions for wall thickness, outside or inside diameter, and material specifications of the nozzle, weld, safe end and pipe. The as-built dimensions are preferable; however, the design dimensions are acceptable. If a safe end is joined with the nozzle, provide its width and the center to center distance between the alloy 82/182 weld and the safe end-to-pipe weld. Provide a detailed configuration drawing at Location 6 also.

   OPPD Response

   The drawings are proprietary to Westinghouse. The requested dimensions are shown below:

   Location 1 (stainless weld):

   t (weld counter bore minimum thickness) = 2.6875"

   OD (outer diameter) = 38.5"

   ID (inside diameter) = 33.125"

   Location 1 (Alloy 82/182 weld):

   t = 3.0313"

   OD = 38.5"

   ID = 32.4375"

   Location 6 (stainless steel):

   t = 1.938"

   OD = 29"

   ID = 25.124"

   Material Type: Pipe = A 451 CPF8M; Elbow = A 351-65 CF8M; Safe end = SA 182 F316; RV Nozzle = A 508-64 Cl. 2

2. Please discuss the differences in the parameters used that resulted in two sets of crack sizes at Location 1 (assuming the same methodology was used).

   OPPD Response

   Differences in the parameters used are; pipe wall thickness, weld type, material properties.

3. Please explain why the leakage and critical crack sizes derived for the Alloy 82/182 weld at Location 1 were not included in Tables 6-1, 7-1, 7-2 of WCAP-17262-P as part of analytical results.

OPPD Response

The information was not included in Tables 6-1 and 7-2 since they are proprietary however; the information was included at the bottom of Tables 6-1 and 7-2. Table 7-1 shows the J-integral analysis results of the cast piping and J-integral is not applicable for Alloy 82/182 weld for which LIMIT load method applies.

NRC RAI 5

The NRC staff plans to perform an independent calculation and requests the following information for Locations 1 and 6 that were analyzed in WCAP-17262-P:

1. Please provide the primary membrane stress and bending stress for Locations 1 and 6.

OPPD Response

At Location 1, normal primary membrane stress=6.027 ksi and normal bending stress= 8.406 ksi, at Location 1 faulted primary membrane stress= 6.495 ksi and faulted bending stress= 9.401 ksi.

At Location 6, normal primary membrane stress= 6.185 ksi and normal bending stress= 2.613 ksi, at Location 6 faulted primary membrane stress= 6.527 ksi and faulted bending stress= 4.178 ksi.

2. For the Alloy 82/182 weld in Location 1, discuss whether the weld residual stresses were included in the calculations. If yes, provide the weld residual stresses. If not, provide justification.

OPPD Response

Weld overlay was not applied for the Alloy 82/182 weld in Location 1 and therefore, weld residual stresses were not included in the leak rate calculations.

3. Please provide Ramberg-Osgood exponents (n). and coefficient (alpha) for the pipe and Alloy 82/182 weld materials, if Ramberg-Osgood equation was used in the flaw evaluations.

OPPD Response

For the J-integral stability analyses at pipe weld Locations 1 and 6 Ramberg-Osgood exponents (n) and coefficient (alpha) were used and the values are: Location 1, n=6.6073 and alpha=0.0213; Location 6, n=6.3819 and alpha=0.287. For the Alloy 82/182 weld in Location 1, Ramberg-Osgood exponents (n) and coefficient (alpha) were not used in the LIMIT load stability analysis.

4. Please discuss whether the fluid is saturated or subcooled before exiting the leakage crack.

   OPPD Response

   The fluid is subcooled before exiting the leakage crack.

NRC RAI 6

Section 2 of WCAP-17262-P discusses operating experience of stress corrosion cracking and fatigue without providing the operating experience of the RCS primary loop piping at Fort Calhoun. Please provide any prior occurrences of fatigue cracking or PWSCC cracking in the RCS primary loop piping.

OPPD Response

In October 2000, Fort Calhoun Station experienced leakage at a pressurizer instrument nozzle (i.e., TE-108). The leakage occurred at an Alloy 82/182 j-groove weld, and was dispositioned as PWSCC. The weld was repaired with a partial penetration Alloy 52/152 weld. No additional leakage occurred in any other pressurizer nozzles and the pressurizer and its nozzles were replaced during the 2006 refueling outage with stainless steel nozzles and welds. This pressurizer nozzle crack is the only known occurrence of PWSCC in RCS primary loop piping at Fort Calhoun Station.

The steam generators and the reactor vessel head were also replaced during the 2006 refueling outage. These new components contain Alloy 52/152 welds, which will significantly decrease the incidence of PWSCC in them and at the associated RCS nozzle to piping locations. Fort Calhoun Station has not experienced any occurrences of fatigue cracking in the RCS primary loop piping, and is compliant with the requirements of EPRI MRP-146, which manages thermal fatigue in non-isolable RCS lines.

NRC RAI 7

In selecting the critical location for analysis, the licensee used axial forces, moments and stresses in Tables 3-1 and 3-2 in WCAP-17262-P. The licensee selected Locations 1 and 6 to perform the flaw analysis. Based on values in Tables 3-1 and 3-2 it appears that Locations 3 and 4 have higher loads and stresses than that of Location 6. Please discuss why Locations 3 and 4 were not also selected for the flaw analysis.

OPPD Response

Locations 3 and 4 are located in the hot leg and Location 1 in the hot leg governs with higher faulted stress than Locations 3 and 4. Location 6 is selected from the cold legs and cross-over legs with highest faulted stress amongst all the weld locations (Locations 5 to 14) in the cold legs and cross-over legs.