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RS-12-074

10 CFR 50.55a

April 13, 2012

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Quad Cities Nuclear Power Station, Unit 2
Renewed Facility Operating License No. DPR-30
NRC Docket No. 50-265

Subject: Response to Request for Additional Information Regarding Relief Request I4R-19, Corrosion and Flaw Evaluations

References:

1) Letter from P. R. Simpson (Exelon Generation Company, LLC) to NRC, "Relief Request I4R-19 Associated with the Reactor Pressure Vessel Nozzle Repairs," dated April 6, 2012

2) Letter from D. M. Gullott (Exelon Generation Company, LLC) to NRC, "Response to Request for Additional Information Regarding Relief Request I4R-19 Associated with the Reactor Pressure Vessel Nozzle Repairs," dated April 12, 2012

3) Email from Joel Wiebe (NRC) to D. M. Gullott (Exelon Generation Company, LLC), "Quad Cities Unit 2 Relief Request I4R-19 – Followup Question to Response to RAI Question 8," dated April 13, 2012

4) Email from Joel Wiebe (NRC) to D. M. Gullott (Exelon Generation Company, LLC), "Quad Cities Unit 2 Relief Request I4R-19 – Followup Question to Regarding the Flaw Evaluation," dated April 13, 2012

In Reference 1, in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC), requested NRC approval of a relief request associated with the Fourth Inservice Inspection (ISI) Interval for Quad Cities Nuclear Power Station (QCNPS), Unit 2. Note that the fourth interval of the QCNPS ISI program complies with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1995 Edition with addenda through 1996.

During review of the subject relief request, the NRC concluded that additional information would be needed to complete their review. In Reference 2, EGC provided a portion of the requested information. In Reference 3, the NRC requested follow-up information to supplement the information provided in Reference 2 regarding the Corrosion Evaluation (i.e., the information provided in Request for Additional Information (RAI) Question 8). In Reference 4, the NRC requested follow-up information to supplement the information provided in Reference 2 regarding the Flaw Evaluation (i.e., the information provided in RAI Question 9). The follow-up information requested in References 3 and 4 is provided in Attachment 1 to this letter.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Mr. Joseph A. Bauer at (630) 657-2804.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information Regarding Relief Request I4R-19, Corrosion and Flaw Evaluations

ATTACHMENT 1

Response to Request for Additional Information Regarding Relief Request I4R-19 Corrosion and Flaw Evaluations

In Reference 1, in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC), requested NRC approval of a relief request associated with the Fourth Inservice Inspection (ISI) Interval for Quad Cities Nuclear Power Station (QCNPS), Unit 2. Note that the fourth interval of the QCNPS ISI program complies with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1995 Edition with addenda through 1996.

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Corrosion Evaluation RAI 1

With respect to general and galvanic corrosion of low-alloy steel, the licensee's corrosion evaluation, provided as Attachment 2 to Letter RS-12-069 dated April 12, 2012, references "NECD-21120, "Monticello Feedwater Nozzle Cladding Crack Repair Report," as the basis for low general corrosion rates and the absence of galvanic corrosion. The licensee's corrosion evaluation references "D.C. Vreeland, et. Al., "Corrosion of Carbon and Low-Alloy Steels in Out-of-Pile Boiling Water Reactor Environment," Corrosion, Vol. 17, No. 6, 1961, as the basis for minimal crevice corrosion of low alloy steel in a boiling water reactor (BWR) operating environment.

Requested Information

To ensure these two references provide bounding corrosion rates for Quad Cities, Unit 2, compare the chemistry of the steam or water to which the low-alloy steel in the repaired N-11B nozzle will be exposed (in the gap between the nozzle remnant and new nozzle) to the chemistry used in the tests described in the two references discussed above. The specific parameters to be addressed should include the concentration of dissolved oxygen and impurities such as chloride and sulfate, conductivity, hydrogen, electrochemical corrosion potential (ECP), and any other parameters that may affect corrosion rates. Values of the parameters for Quad Cities may be typical or average values or maximum values. If the steam/water environment tests documented in the references do not envelop or bound the actual conditions to which the low-alloy steel in Nozzle N-11B will be exposed, provide a justification for the use of these reference to support the prediction of minimal corrosion for the exposed low-alloy steel, or provide additional references.

Response

The corrosion evaluation (51-9180975-001) references NEDC-21120 regarding general and galvanic corrosion. The corrosion rate from NEDC-21120 is based on tests performed with low alloy steel in contact with BWR water conditions and BWR steam conditions. The corrosion rate used in 51-9180975-001 is 70% faster than the maximum corrosion rate than all of the tests from NEDC-21120. NEDC-21120 does not provide specific water chemistry parameters. NEDC-21120 was written in 1975; therefore, it can be inferred that the BWR water chemistry used in the tests was consistent with the water chemistry requirements for BWRs of 1975 or before. The water chemistry requirements of this time period are known to be much less stringent than today's modern requirements (i.e., BWRVIP-130 and more recently BWRVIP-190). BWRVIP-130 was created to better control BWR water chemistry and is significantly more restrictive than the requirements of the 1970s (and before). Quad Cities Unit 2 reactor water meets or exceeds the requirements of BWRVIP-190, which supersedes BWRVIP-130. BWRVIP-190 has requirements that are equal to or more restrictive than BWRVIP-130. Based on this information, it is concluded that the water chemistry at Quad Cities Unit 2 is less aggressive than the water chemistry used in the tests discussed in NEDC-21120.

The corrosion evaluation (51-9180975-001) references Vreeland (1961) regarding crevice corrosion. For the same reasoning given above, it is concluded that the water chemistry at Quad Cities Unit 2 is less aggressive than the water chemistry used in the tests discussed in Vreeland (1961).

Flaw Evaluation RAI-1

The flaw evaluation report states that heatup and cooldown, SRV blowdown, and SCRAM are determined to be controlling for the current flaw evaluations. Please provide the criteria and the qualitative assessment that you have performed to determine that SRV blowdown and SCRAM are the controlling emergency/faulted and normal/upset transients among all design transients.

Response

The bounding transients were determined from the Section III design analysis of record. The transients that have significant temperature excursions, and therefore, significant stress ranges, are heatup and cooldown, SCRAM, and SRV Blowdown. Therefore, these are the transients that lead to the highest stress ranges, and they are controlling for the flaw evaluation.

Flaw Evaluation RAI-2

Confirm that the crack plane is assumed to exist at a circumferential location with respect to the nozzle hole such that the maximum remote membrane and bending stresses in Reference 5 were used for the subsequent fracture mechanics analysis. Justify your selection of the crack plane location if this is not the case.

Response

Yes, the crack plane is assumed to exist at a circumferential location with respect to the nozzle hole such that the maximum remote membrane and bending stresses are applied. The stresses used are normal to the axial plane of the vessel, where the hoop stresses are at a maximum.

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Flaw Evaluation RAI-3

Considering the difference in Young's modulus (E) values of the low alloy steel and the 182 weld and that the "local" weld residual stresses may not be adequately represented by the "averaging" membrane and bending stresses, please demonstrate that your approach will not severely underestimate the effect of residual stresses on the applied stress intensity factor.

Response

The difference in Young's modulus between the Alloy 182 weld material and the SA-302B (modified) low alloy steel (LAS), is reasonably small, and is not expected to have any significant effect on the material behavior in the elastic regime. Weld residual stresses are steady state secondary stresses, which are generally limited by yield strength. The Alloy 182 and LAS materials have similar yield strengths and the difference will not significantly affect the weld residual stress distribution. Since the vessel and 3/16 inch of the original weld were stress relieved during original fabrication, weld residual stresses were significantly relieved in those materials. The rewelding of the J-groove weld produced tensile residual stresses in the weld at roughly the yield level at temperature, and for equilibrium, the stresses in the adjacent LAS material will be slightly compressive and the stresses in the original Alloy 182 weld will be further relieved, possibly to the point of being compressive. In the present analysis, the crack tip is assumed to be within the LAS material, so the residual stresses assumed are conservative for the present analysis. Furthermore, assuming an average compressive stress in the LAS material will underestimate the compressive stresses near the crack front, resulting in a conservative analysis.

Flaw Evaluation RAI-4

Section 2.1 of the flaw evaluation report described the Raju-Newman stress intensity factor model, but Section 3.3's description is not clear: "The SIF solution used is based on a corner flaw in a flat plate under remote bending and tension." Please confirm that the flaw evaluation is based on the Raju-Newman model without any further simplifications.

Response

Yes, the flaw evaluation is based on the Raju-Newman model without any further simplifications. Additionally, stresses due to weld residual stresses are estimated and superimposed with the stresses due to operating transients. The analysis also considers the applicable pressure on the crack face at the particular transient time point being evaluated.

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Corrosion and Flaw Evaluations**

Flaw Evaluation RAI-5

Section 4.2 specified a certain crack depth increment to account for any potential flaw growth without performing an actual flaw growth analysis. Please justify this assumed crack depth increment by providing an estimated crack growth within one cycle using appropriate fatigue crack and stress corrosion cracking growth rates (BWRVIP-60), albeit it is a small value.

Response

The stress corrosion cracking (SCC) growth rate in the steam space region where the instrument nozzle N-11B is located is expected to be negligible as discussed on Page 5-5 of BWRVIP-60A. Fatigue crack growth is also considered to be negligible over a two year period. Conservatively, per BWRVIP-60A, Page 8-1, the upper bound crack growth rate in the low alloy steel shell provided is 2.83×10^{-6} in/hr. This rate is applicable for the BWR water environment at 550 °F which corresponds to the steady state operating condition. For the next two year fuel cycle period, this corresponds to 0.0496 inches of stress corrosion crack growth. Hence, a value of 1/16" or 0.0625 inches is conservative and allows for the potential growth rate due to SCC and fatigue in the low alloy steel vessel material.

REFERENCES

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