ATTACHMENT 6

Structural Integrity Associates, Inc.® Report

File No. 1400187.302, Revision 2

Probability of Failure for LaSalle Unit 2 N1
Nozzle-to-Shell-Welds and Nozzle Blend Radii Regions

(Non-Proprietary)

13 pages follow

Structural Integrity Associates, Inc.®

CALCULATION PACKAGE

File No.: 1400187.302

Project No.: 1400187

Quality Program: ☒ Nuclear ☐ Commercial

PROJECT NAME:
LaSalle N702 Relief Request for 60 Years

CONTRACT NO.:
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CLIENT:
Exelon Generation Company LLC

PLANT:
LaSalle County Generating Station, Units 1 and 2

CALCULATION TITLE:
Probability of Failure for LaSalle Unit 2 N1 Nozzle-to-Shell-Welds and Nozzle Blend Radii Regions

NOTE: This document contains ***vendor proprietary information****. Such information has been redacted for public release of this document.*

Document Revision	Affected Pages	Revision Description	Project Manager Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1 - 11 A-1 - A-2	Initial Issue	Wilson Wong 8/8/14	Jim Wu 8/8/14 Wilson Wong 8/8/14
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1.0 INTRODUCTION

Structural Integrity Associates (SIA) is contracted by Exelon to perform a plant specific analysis to request inspection relief for the current licensing period per ASME Boiler and Pressure Vessel Code Case N-702 [1], and to the end of the period of extended operation (60 years of operation) for the LaSalle County Generating Station (LGS) RPV nozzles. LaSalle intends to extend their existing relief request for multiple RPV nozzles, specifically the N1, N2, N3, N5, N6, N7, N8, N9, N16, and N18 nozzles. [redacted] It is also stated in Section 2 of Reference 2 that "It should be noted that only the recirculation inlet and outlet nozzles need to be checked because the P(FlE)s (Conditional probability of failure from event F due to event E) for other nozzles are an order of magnitude lower." SIA concluded that the N1 nozzle of Unit 2 is the bounding nozzle since it is the only nozzle violating the condition 4 requirements set forth by BWRVIP-241 [2]. To address the elevated fluence issue of certain nozzles in the belt-line region of the RPV, a bounding approach is used to qualify all of the indicated nozzles for both units by analyzing the Unit 2 N1 nozzle using the fluence level from the N6 nozzle (peak fluence at end of the period of extended operation) since N6 nozzle is located in the belt-line region and has the bounding fluence among all indicated nozzles.

The intent of this analysis is to confirm that the N1 nozzles meet the applicable acceptance criteria considering the elevated fluence level, thus qualifying all nozzles identified above. The evaluation consists of two parts: Finite Element Model (FEM) Stress Analysis and Probabilistic Fracture Mechanics (PFM) Analysis. This calculation package documents the PFM analysis while a previous calculation package [8] documented the stress analysis.

2.0 METHODOLOGY

The approach used for this evaluation is consistent with the methodology presented in Reference 3 and 5. A Monte Carlo simulation is performed using a variant of the program VIPER [4] with some modifications as described in the following sections. The VIPER program was developed as part of the program in Reference [3] for the Boiling Water Reactor (BWR) reactor pressure vessel (RPV) shell weld inspection recommendations. The software was modified into a separate edition, identified as VIPERNOZ, for use in this evaluation.

The detailed description of the methodology incorporated in the VIPER/VIPERNOZ program is documented in References [3] and [5].

3.0 SOFTWARE MODIFICATIONS

Several modifications were made to VIPER in order to include the capability to perform the evaluation for nozzle blend radii. The modifications are:

1. Include fatigue crack growth analysis,
2. Option to perform stress corrosion crack growth and/or fatigue crack growth,

3. User defined flaw size distribution,
4. User defined probability of detection (PoD) curves for inspection,
5. User defined event occurrence time,
6. User defined distribution for selected random parameters,
7. User input number of printout for failed and non-failed vessels,
8. The constant for margin term for upper bound values of adjusted reference temperature required by Appendix G to 10 CFR Part 50 is a user input,
9. Pre-service inspection is eliminated,
10. Initial flaw size to include clad thickness is a user option,
11. Improvement in data structure for analysis results.

The modified software for this project is identified as VIPERNOZ to distinguish from the original VIPER software in Reference [3]. Note that the VIPERNOZ computer program is the same program used in the BWRVIP-108NP report that was accepted by the NRC in their SER [5].

4.0 ASSUMPTIONS

The following assumptions used in the evaluation are consistent with those listed in References [2] and [5]:

[redacted]

3. The flaw size distribution, PVRUF, is assumed to be as shown in Figure 5-4 of Reference [6].

[redacted]

5. Lower bound constant upper shelf fracture toughness is set to 200 ksi√in with a standard deviation of 30 ksi√in for un-irradiated material based on the SER report. For irradiated material the VIPERNOZ program will make the necessary adjustment based on fluence and Initial RT_{NDT} inputs using guidance from RG 1.99 [12].
6. Standard deviation of the mean K_{IC} is set to 15 percent of the mean value of the K_{IC} per the SER report [5].
7. All chemistry information from N1 nozzle-to-shell weld and nozzle blend radii was conservatively taken from BWRVIP-241 fleet bounding data. [2]
8. Peak fluence from the Unit 2 N6 nozzles at the belt line region will conservatively be used for the N1 nozzle-to-shell weld and nozzle blend radii.

5.0 DESIGN INPUT

The LaSalle plant specific input is described below.

- Vessel Wall Thickness at the weld = 6.5625" (excluding clad) [7]
- Vessel Wall Thickness through the Blend = 10.4626" (excluding clad) [Path 3, 8]
- Vessel Wall Thickness through the Blend = 11.2653" (excluding clad) [Path 1, 8]
- Vessel Inner Radius = 126.6875" (excluding clad) [7]
- Vessel Clad Thickness at Blend = 0.1875" [7]
- Vessel Clad Thickness at Weld = 0.1875" [7]
- Vessel Operating Temperature = 528°F [9]
- Vessel Hydro Testing Temperature = 100°F [9]
- Operating Pressure = 1050 psig [9]
- Pressure during Bounding Transient = 1180 psig [9]
- End of Life Fluence (54 EFPY/60 years) for N6 Forging at Unit 2
  = 5.36×10^{17} n/cm^2 [Table 7-9, 11]
- Mean Initial RT_{ndt} / standard deviation at Blend Radius
  = [redacted]
- Mean Initial RT_{ndt} / standard deviation at the RPV Weld
  = [redacted]

The weld chemistry is taken from Reference 2 and presented in Table 1.

All random variables are summarized in Table 2 of Reference [12]. Most of the input is obtained from Reference [3], except standard deviation for %Cu and %Ni for nozzle blend radii and nozzle-to-shell weld. For nozzle blend radii, these inputs are equal to 0.04407 (Calculated based on Figure 3-1 and 3-2 of Reference 16) and [redacted]

6.0 FATIGUE CRACK GROWTH

The fatigue data for SA-533 Grade B Class 1 and SA-508 Class 2 in a reactor water environment are reported in Reference [13] for weld metal testing at R = 0.2 and 0.7. To produce a fatigue crack growth law and distribution for the VIPERNOZ software, the data for R= 0.7 was fitted into a form of Paris Law. The R= 0.7 fatigue crack growth law was chosen for conservatism. The curve fit results of the mean fatigue crack growth law is presented with the Paris Law shown as follows:

$$\frac{da}{dn} = 3.817 * 10^{-9} (\Delta K)^{2.927} \quad (1)$$

where a = crack depth, in
n = cycle
$\Delta K = K_{max} - K_{min}$, ksi-in$^{0.5}$

A comparison to the ASME Section XI [10] fatigue crack growth law in a reactor water environment is documented in Reference [13]. It shows a reasonable comparison where the ASME Section XI law is more conservative on growth rate at high ΔK.

Using the rank ordered residual plot, it was shown that a Weibull distribution was more representative for the data. The Weibull residual plot with the linear curve fit of the data is shown below:

$$y = -0.3712 + 4.15x \quad (2)$$

where $y = \ln(\ln(1/(1-F))$
$x = \ln((da/dn)_{actual}/(da/dn)_{mean})$
F = cumulative probability distribution

Per 10CFR 50.55a, the NRC have placed additional, more limiting requirements on the Section XI fatigue crack growth (FCG) in Appendix A for negative R ratios. Since Reference 14 has concluded that the main contributing factor of crack growth is SCC and that fatigue crack growth is negligible, the effect of FCG due to negative R ratios need not be addressed.

7.0 STRESS RESULTS AND FATIGUE CYCLE LOADINGS

The stress analyses for the nozzle-to-shell weld and the nozzle blend radius for the Unit 2 N1 nozzle are presented in Reference [8]. The stress analyses were performed for unit pressure and bounding normal and upset thermal transients (Loss of Feedwater Pumps/Isolation Valves Close) for the N1 nozzle. The azimuthal locations evaluated were 0° and 90°, which also represent the symmetric un-modeled 180° and 270° locations of the nozzle. Two through-wall sections were selected. One is at the location of the weld between the RPV and nozzle and the other is at the blend radius location of the nozzle.

The bounding load cases analyzed for the N1 nozzle include:

1. Unit pressure
2. Turbine Generator Trip-SCRAM (TGT-SCRAM)
3. Loss of Feedwater Pumps/Isolation Valves Close

For the thermal transients, the through-wall stress profiles that produce the largest stress ranges for thermal fatigue crack growth are presented and used in the evaluation.

The number of thermal cycles for the TGT-SCRAM transient are considered to be the total number of cycles for all normal and upset conditions that involve temperature/ pressure changes in region B of the reactor vessel (754 cycles per Reference 9 for 40 years of operation and approximately 1131 cycles for

60 years of operation, or 189 cycles for each block of 10 years of operation) for conservatism. Specifically, transients considered were: Design Test (130 cycles), Start Up (117 Cycles), Loss of Feedwater Heater (80 Cycles), SCRAM (180 Cycles), Shut Down Vessel Flooding (111 Cycles), Unbolt (123 Cycles), Loss of Feedwater Pump/Isolation Valves Close (10 Cycles), and Natural Circulation Start Up (3 Cycles).

The number of thermal cycles for the Loss of Feedwater Pump/Isolation Valves Close transient is 10 cycles for 40 years of operation per Reference 9. However, there are three internal cycles within the main transient, the last of which occurs after an indefinite time and can be bounded by the TGT-SCRAM transient. Therefore, only the first two internal cycles are considered for the Loss of Feedwater Pump/Isolation Valves Close transient, which amounts to 20 cycles for 40 years of operation (10 cycles × 2 internal cycles) and 30 cycles for 60 years of operation.

8.0 PROBABILISTIC FRACTURE MECHANICS EVALUATION

The probabilistic evaluation is performed for the case of 25% inspection for the extended operating period (with zero inspection coverage conservatively assumed for the initial 40 years of operation).

For the nozzle blend radius region, a nozzle blend radius crack model [15] was used in the probabilistic fracture mechanics evaluation for the reliability of the in-service inspection program. For this location and crack model, the applicable stress is the stress perpendicular to any path cut along the nozzle longitudinal axis (nozzle hoop stress).

For the nozzle-to-vessel shell weld, either a circumferential or an axial crack could be initiated due to either component fabrication (i.e. considering only welding process) or stress corrosion cracking. From Reference [3], it is shown that the probability of failure for a circumferential crack is much less than an axial crack, due to the difference in the stress (hoop versus axial) and the influence function of the crack model. Therefore, the probabilistic fracture mechanics evaluation for the nozzle and vessel shell weld would concentrate on the axial crack. An axial elliptical crack model with a crack aspect ratio of a/l = 0.2 is used in the evaluation for the nozzle-to-vessel shell weld. The inspection PoD curve is the user input of Figure 42 of Reference [12], with an inspection interval every 10 years. The calculation of stress intensity factor is at the deepest point of the crack.

The analyses are performed using VIPERNOZ, a modified version of the program VIPER, [4], with the modifications as described in Section 3.0. The number of simulations is 5 million.

9.0 RESULTS OF ANALYSES

The reliability evaluation is presented using plant specific inspection coverage. The probabilities of failure (PoF) from the limiting Low Temperature Overpressure (LTOP) events and Normal Operating Conditions are summarized in Table 2. The in-service inspection of 25% inspection for the extended operating term (with zero inspection coverage for the initial 40 years of operation) is used at both the nozzle blend radius as well as the nozzle-to-shell weld.

10.0 CONCLUSIONS

The probability of failure per reactor year for the nozzle-to-shell-weld and nozzle blend radii in the limiting N1 nozzle at LaSalle Unit 2 is below the criteria of 5×10^{-6} per year [17]. The LaSalle N1 nozzles still meet the acceptable failure probability considering 60 year thermal cycles and theelevated fluence level of the N6 nozzles. Therefore, N1, N2, N3, N5, N6, N7, N8, N9, N16, and N18 nozzles at LaSalle Units 1 and 2 still qualify for reduced inspection using ASME Code Case N-702 to the end of the period of extended operation (60 years of operation).

11.0 REFERENCES

1. Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1," February 20, 2004.
2. *BWRVIP-241: BWR Vessel Internal Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii*, EPRI, Palo Alto, CA. 1021005. **EPRI PROPRIETARY INFORMATION.**
3. BWRVIP Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Electric Power Research Institute TR-105697, September 1995. **EPRI PROPRIETARY INFORMATION.**
4. VIPER, Vessel Inspection Program Evaluation for Reliability, Version 1.2 (1/5/98), Structural Integrity Associates.
5. Safety Evaluation of Proprietary EPRI Report, "BWR Vessel and Internal Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)," December 19, 2007.
6. *BWRVIP-108NP: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii.* EPRI, Palo Alto, CA: 2007. 1016123
7. GE Drawing, "Recirculation Outlet Nozzle N1," LaSalle II MPL# B13-D003, SI File No. 1400187.202.
8. SI Calculation Package, "Finite Element Model Development and Thermal Mechanical Stress Analyses for the Unit 2 N1 Nozzle," Revision 0, SI File Number 1400187.301.
9. Thermal Cycle Diagrams
    a. General Electric Drawing Number 158B8136, Sheet 1, Revision 6, "Reactor Vessel Nozzle Thermal Cycles," SI File No. 1400187.207
    b. General Electric Drawing Number 731E776, Sheets 1 and 2, Revision 3, "Reactor Vessel Thermal Cycles," LaSalle Unit 1, SI File No. 1400187.205
    c. General Electric Drawing Number 761E581, Sheets 1 and 2, Revision 1, "Reactor Vessel Thermal Cycles," LaSalle Unit 2, SI File No. 1400187.206.
10. ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, 2007 Edition with 2008 Addenda.
11. EXL-LSA-001-R-003, "LaSalle County Generating Station Unit 2 Reactor Pressure Vessel Fluence Evaluation at End of Cycle 15 with Projections to 32 and 54 EFPY," Revision 0, SI File Number 1400187.209.
12. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988..
13. Bamford, W. H., "Application of corrosion fatigue crack growth rate data to integrity analyses of nuclear reactor vessels," Journal of Engineering Materials and Technology, Vol. 101, 1979.

14. EPRI Memo 2012-138, "BWRVIP Support of ASME Code Case N-702 Inservice Inspection Relief," From Chuck Wirtz to All BWRVIP Committee Members, August 31, 2012.

15. ASME publication, "Fracture Mechanics Analysis of JAERI Model Pressure Vessel Test," S.A. Delvin and P C. Ricardella, 78-PVP-91. .

16. BWRVIP-173-A: "Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials," EPRI, Palo Alto, CA, 2011, 1022835, SI File Number BWRVIP-173-A. **EPRI PROPRIETARY INFORMATION.**

17. Technical Basis for Revision of Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61), NUREG-1806, Vol. 1, August 2007.

Table 1: LaSalle Weld Chemistry

	Mean Chemistry	
	%Cu	%Ni
N1 Nozzle-to-shell-weld	[redacted]	[redacted]
N1 Nozzle Forging Blend Radius	[redacted]	[redacted]

Note: %Cu and %Ni were obtained from Reference 2, Table 5-1.

Table 2: Probability of Failure Results Summary

	PoF per year from LTOP events for 25% In-Service Inspection for period of Extended Operation (Zero inspection for initial 40 years)*	PoF per year from Normal Operating Condition for 25% In-Service Inspection for period of Extended Operation (Zero inspection for initial 40 years)	Maximum PoF per year [17]
Nozzle Blend Radii	1.4×10^{-9}	4.2×10^{-7}	5.0E-6
Nozzle-to-shell-weld	$<<2.0 \times 10^{-10}$	3.3×10^{-9}	5.0E-6

*Note: Values include 1×10^{-3} probability of LTOP event occurrence.

APPENDIX A

LIST OF SUPPORTING FILES

File Name	Description
LCNS_Blend_p1.INP	VIPERNOZ input file for Path 1 at nozzle blend radii.
LCNS _Blend_p3.INP	VIPERNOZ input file for Path 3 at nozzle blend radii.
LCNS _Weld_p2.INP	VIPERNOZ input file for Path 2 at nozzle-to-shell-weld.
LCNS _Weld_p4.INP	VIPERNOZ input file for Path 4 at nozzle-to-shell-weld.
LCNS _Blend_p1.OUT	VIPERNOZ output file for Path 1 at nozzle blend radii.
LCNS _Blend_p3.OUT	VIPERNOZ output file for Path 3 at nozzle blend radii.
LCNS _Weld_p2.OUT	VIPERNOZ output file for Path 2 at nozzle-to-shell-weld.
LCNS _Weld_p4.OUT	VIPERNOZ output file for Path 4 at nozzle-to-shell-weld.
VIPERNOZ_v2.EXE	VIPERNOZ executable program
ISPCTPOD.EXE	VIPERNOZ probability of detection curve input file
FLWDSTRB.EXE	VIPERNOZ flaw size distribution curve input file