Attachment 2

To PNPS Letter 2.15.073

Entergy Calculation M1396,
***Structural Integrity Associates*, "*Evaluation of Probability of Failure for Recirculation* Inlet (N2) in the Nozzle-to-Shell Welds and Nozzle Blend Radii Regions at Pilgrim Nuclear Station," 1400071.301, Revision 0, February 2014.**

(15 Pages)

ATTACHMENT 9.2 ENGINEERING CALCULATION COVER PAGE

Sheet 1 of 2

☐ ANO-1 ☐ ANO-2 ☐ GGNS ☐ IP-2 ☐ IP-3 ☐ PLP
☐ JAF ☒ PNPS ☐ RBS ☐ VY ☐ W3
☐ NP-GGNS-3 ☐ NP-RBS-3

CALCULATION COVER PAGE	(1) **EC # 0000049395**	(2) **Page 1 of 4**

(3) Design Basis Calc. ☐ YES ☒ NO | (4) ☒ CALCULATION ☐ EC Markup

(5) **Calculation No: M1396**	(6) **Revision: 0**
(7) **Title: Evaluation of the Probability of Failure for Recirculation Inlet (N2) Nozzles (SI 1400071.301)**	(8) **Editorial** ☐ YES ☒ NO
(9) **System(s): 54**	(10) **Review Org (Department): DESMECH**

(11) **Safety Class:** ☒ **Safety / Quality Related** ☐ **Augmented Quality Program** ☐ **Non-Safety Related**	(12) **Component/Equipment/Structure Type/Number:** **N2 Nozzles**	
(13) **Document Type: B4.01**		
(14) **Keywords (Description/Topical Codes):** **RPV** **N2 Nozzle Evaluation**		

REVIEWS

(15) Name/Signature/Date SIA (See SIA Cover Sheet **Responsible Engineer**	(16) Name/Signature/Date K E Woods/ [signature] / 2-28-14 ☐ **Design Verifier** ☒ **Reviewer** ☒ Comments Attached	(17) Name/Signature/Date R M Pace / [signature] 3/5/14 **Supervisor/Approval** ☐ Comments Attached

Comments Calculation M1396 Rev 0

The attached Structural Integrity Associates (SIA) prepared and independently verified calculation 1400071.301 Rev 0 has 11 pages. The calculations was prepared under the Entergy approved SIA 10CFR50 Appendix B QA Program.

Total pages including 4 pages for Entergy comments, and Attachments 9.2, 9.3, and 9.4 are 15.

The following comments were provided to SIA after receiving their draft Calculation and have been adequately addressed:

1. Most of the Design Input comes from Reference 8, SI Calculation 0801317.302 "Pilgrim Recirculation Inlet Probability Fracture Mechanics Analyses". What project is this from? I cannot find this calculation in our database, is this something you can share with us? Also, I cannot find SI Calculation 0801317.301, Pilgrim Recirculation Inlet Nozzle Stress Analysis, [Reference 13] in our database either.
2. Section 5.0, bullet for End of Life Fluence (56 EFPY/60 Years) should be **54 EFPY**
3. Section 8.0, the first sentence reads, "The stress analyses for the nozzle-to-shell weld and the nozzle blend radius for the N2 nozzle are presented in Reference [14]." This should be **Reference [13].**
4. Section 8.0, should add a discussion of beyond 40 years of operation and 245 allowed cold startups for 60 years (or 287 cold startups allowed for 80 years). I have attached our most recent Cycle Count spreadsheet for transients.
5. In BWRVIP-241 Table 5-2 contains a count of transients for Pilgrim Station, Loss of Feedwater Pumps has 10 in the table. We are currently at 13. A discussion about this should be added.
6. Section 8.0, why is Reference 15 used for the discussion of transients? Consider using SI Calculation 0800533.32, 60-Year and 80-Year Plant Cyclic Duty Projection or SI Calculation 0800533.310, Thermal Cycle Counts.
7. Consider adding SI Calculation 0800533.32, 60-Year and 80-Year Plant Cyclic Duty Projection and SI Calculation 0800533.310, Thermal Cycle Counts to the references.

CALCULATION REFERENCE SHEET	CALCULATION NO: M1396 REVISION: 0

I. EC Markups Incorporated (N/A to NP calculations)
1. N/A
2.
3.
4.
5.

II. Relationships:	Sht	Rev	Input Doc	Output Doc	Impact Y/N	Tracking No.
1. ISI Program, N2 Nozzle Inspection			☐	☒	Y	Relief Request PRR-24
2.			☐	☐		
3.			☐	☐		
4.			☐	☐		
5.			☐	☐		

III. CROSS REFERENCES:
1. SI Calculation 1400071.301, Rev 0
2.
3.
4.
5.

IV. SOFTWARE USED:

Title: VIPER Version/Release: 1.2 Disk/CD No. N/A

V. DISK/CDS INCLUDED:

Title:________ Version/Release________ Disk/CD No.______

VI. OTHER CHANGES:

Revision	Record of Revision
0	Initial issue.

Structural Integrity Associates, Inc.®

CALCULATION PACKAGE

File No.: 1400071.301

Project No.: 1400071

Quality Program: ☒ Nuclear ☐ Commercial

PROJECT NAME:
Pilgrim N-702 Evaluation

CONTRACT NO.:
10404807

CLIENT:	**PLANT:**
Entergy Nuclear	Pilgrim Nuclear Station

CALCULATION TITLE:
Evaluation of the Probability of Failure for Recirculation Inlet (N2) in the Nozzle-to-Shell-Welds and Nozzle Blend Radii Regions at Pilgrim Nuclear Station

Document Revision	Affected Pages	Revision Description	Project Manager Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1 - 9 A-1 - A-2	Initial Release	H L Gustin H. L. Gustin, P. E. 2/27/14	**Responsible Engineer:** Jim Wu Jim Wu 2/27/14 **Responsible Verifier:** Wilson Wong Wilson Wong 2/27/14

Table of Contents

List of Tables

1.0 INTRODUCTION

Structural Integrity Associates (SI) is contracted by Entergy to perform a plant specific analysis at Pilgrim Nuclear Station to extend applicability of the existing relief request, performed using the methods of ASME Boiler and Pressure Vessel Code Case N-702 [1], to the end of the period of extended operation (60 years of operation) for the Pilgrim Nuclear Station recirculation inlet nozzle (N2) nozzle-to-vessel welds and nozzle blend radii with an elevated fluence level ($>1\times10^{17}$ n/cm^2). The intent of this analysis is to confirm that the N2 nozzles meet the applicable acceptance criteria considering the elevated fluence level.

2.0 OBJECTIVE

The objective of the evaluation documented in this calculation package is to perform a plant specific analysis to extend applicability of the existing relief request to 60 years of operation (54 EFPY) for the Pilgrim N2 nozzles considering an elevated fluence level within the beltline of the vessel shell ($>1x10^{17}$ n/cm^2). This evaluation considers the nozzle-to-vessel-weld and nozzle blend radius on the N2 nozzle per code case N-702 [1], and confirms that the nozzle still meets the acceptable failure probability considering the elevated fluence level.

3.0 METHODOLOGY

The approach used for this evaluation is consistent with the methodology presented in Reference [2]. A Monte Carlo simulation is performed using a variant of the program VIPER [3] with some modifications as described in the following sections. The VIPER program was developed as part of the program in Reference [2] 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 [2] and [4].

4.0 ASSUMPTIONS

The following assumptions are used in the evaluation and are based on References [4] and [5]:

1. One stress corrosion initiation and 0.1 fabrication flaws are assumed per nozzle blend radius.
2. One stress corrosion initiation and 1.0 fabrication flaws are assumed per nozzle/shell weld.
3. The flaw size distribution, PVRUF, is assumed to be as shown in Figure 43 of Reference [5].
4. The weld residual stress distribution at the nozzle/shell weld is assumed to be a cosine distribution through the wall thickness with 8 ksi mean amplitude and 5 ksi standard deviation.
5. The K_{IC} was calculated for both irradiated and un-irradiated material with the valued bounded at 200 ksi$\sqrt{}$in. This value represents the upper bound value, with actual K_{IC} value being the upper bound or lower. Thus constant upper shelf fracture toughness is set to 200 ksi$\sqrt{}$in per Reference 4 and is considered conservative.

6. Standard deviation of the mean K_{IC} is set to 15 percent of the mean value of the K_{IC} per Table 5-1 of Reference 7.

5.0 DESIGN INPUT

The Pilgrim plant specific input is described below.

- Vessel Wall Thickness at the weld = 9.095" [8]
- Vessel Wall Thickness through the Blend = 7.934" [8]
- Vessel Inner Radius (including clad) = 113.406" [8]
- Vessel Clad Thickness at Blend = 0.21875" [8]
- Vessel Clad Thickness at Weld = 0.21875" [8]
- Vessel Operating Temperature = 546°F [8]
- Vessel Hydro Testing Temperature = 88°F [8]
- Operating Pressure = 1035 psi [8]
- Pressure during Bounding Transient = 1150 psi [8]
- End of Life Fluence (54 EFPY/60 years) for N2 nozzle = 2.81×10^{17} n/cm^2 (Table 5 of Reference 6)
- Initial RT_{ndt} at Blend Radius = 0°F (Conservatively taken at upper bound) [9]
- Initial RT_{ndt} at the RPV Weld = -50°F (Conservatively taken at upper bound) [9]

The weld and nozzle chemistry is obtained from References 9 and 12 and presented in Table 1.

All random variables are summarized in Table 2 of Reference [5]. Most of the input is obtained from Reference [2], except standard deviation for %Cu and %Ni for nozzle blend radii. These inputs are equal to 0.0447 and 0.068 for %Cu and %Ni, respectively, and are obtained from BWRVIP-173 [12].

6.0 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 [2].

7.0 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 [10] 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:

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

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

A comparison to the ASME Section XI fatigue crack growth law in a reactor water environment is documented in Reference [5], it shows a reasonable comparison where the 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:

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

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

8.0 STRESS RESULTS AND FATIGUE CYCLE LOADINGS

The stress analyses for the nozzle-to-shell weld and the nozzle blend radius for the N2 nozzle are presented in Reference [13]. The stress analyses were performed for the load cases of pressure, and the representative thermal transients for the N2 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, see Figure 1 of Reference 13.

The load cases analyzed for the nozzle include:

1. Constant Pressure at 1250 psig
2. Heat Up
3. Sudden Pump Start of Cold Recirculation Loop

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

All the transients and corresponding number of cycles obtained from Reference 15 are consistent with what was used in Reference 8. For PFM analysis, the two bounding thermal transients (Heatup and Sudden Pump Start of Cold Recirculation Loop Transient [15]) are chosen based on their stress. All other transients are bounded by the Sudden Pump Start of Cold Recirculation Loop Transients. The number of cycles for Heatup is 120 for a design life of 40 years [15]. The normal shutdown transient (118 cycles) is combined with Normal Startup (120 cycles) to get 120 full cycles. The total number of cycles from all the transients except Heatup are used for Sudden Pump Start of Cold Recirculation Loop transient, which add up to be 290 cycles. In addition, based on industry experience and studies done with EPRI [16], SCC crack growth represents the majority of the crack growth. Reference 16 demonstrated that crack growth due to additional mechanical/thermal fatigue cycles introduced by the extended operation time is insignificant compared to hypothetical stress corrosion crack growth (SCC). Thus, the amount of thermal cycle driven fatigue crack growth due to the extended operation to 60 years is not a controlling factor in the probability of failure of the BWR reactor vessel nozzles.

9.0 PROBABILISTIC FRACTURE MECHANICS EVALUATION

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

For the nozzle blend radius region, a nozzle blend radius crack model [11] 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.

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 [2], 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 $a/l = 0.5$ 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 [5], 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, with the modifications as described in Section 6.0. The number of simulations is 1 million.

10.0 RESULTS OF ANALYSES

The reliability evaluation is presented using plant specific inspection coverage. The probabilities of failure (PoF) are summarized in Table 2. The in-service inspection of 25% inspection for the extended operating term (with 25% 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. The average PoF is 9.67×10^{-7} per year for the nozzle blend radius, and 1.67×10^{-8} per year for the nozzle-to-shell weld, both of which are less than the 5×10^{-6} per year criteria from Reference [14].

11.0 CONCLUSIONS

The probability of failure per reactor year for the nozzle-to-shell-weld and nozzle blend radii in the N2 nozzles at the Pilgrim Nuclear Station is below the criteria of 5×10^{-6} per year. The Pilgrim Nuclear Station N2 nozzles still meet the acceptable failure probability considering the elevated fluence level per ASME Code Case N-702 for the existing relief request.

12.0 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 Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Electric Power Research Institute TR-105697, September 1995.
3. VIPER, Vessel Inspection Program Evaluation for Reliability, Version 1.2 (1/5/98), Structural Integrity Associates.
4. BWRVIP-108NP, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," 1016123, November 2007, SI File Number BWRVIP-108NP.
5. SI Calculation W-EPRI-180-302, "Evaluation of effect of inspection on the probability of failure for BWR Nozzle-to-Shell-Welds and Nozzle Blend Radii Region," Revision 0.
6. SI Calculation PNPS-22Q-302, "N2 Nozzle Evaluation," Revision 0.
7. BWRVIP-241, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,"1021005, October 2010.
8. SI Calculation 0801317.302, "Pilgrim Recirculation Inlet Probability Fracture Mechanics Analyses," Revision 0.
9. SI Report SIR-00-082, PNPS-03Q-402, "Updated Evaluation of Reactor Pressure Vessel Materials Properties for Pilgrim Nuclear Power Station," Revision 0, August 2000.
10. 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.
11. Private Communication, P. M. Besuner (Failure Analysis Associates) to P. C. Riccardella, "Three Dimensional Stress Intensity Factor Magnification Constant for Radial Feedwater Nozzle Cracks," June 1976.
12. BWRVIP-173-A, "Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials," 1022835, July 2011, SI File Number BWRVIP-173A.
13. SI Calculation 0801317.301, "Pilgrim Recirculation Inlet Nozzle Stress Analyses," Revision 0.
14. 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.
15. Combustion Engineering, Inc., "Analytical Report for Pilgrim Reactor Vessel," Report No. CENC-1139, 1979, SI File No. PNPS-03Q-217.
16. Email from Chuck Wirtz (FirstEnergy) to BWRVIP Committee Members, "BWRVIP Support of Code Case N-702 Inservice Inspection Relief." SI File Number 1400071.201.

Table 1: Pilgrim Weld Chemistry

	Chemistry	
	%Cu	%Ni
N2 Nozzle-to-shell-weld	0.18	1.04
N2 Nozzle Forging Blend Radius	0.18	0.85

Table 2: Probability of Failure Results Summary

	PoF for 25% In-Service Inspection for period of Extended Operation (25% inspection for initial 40 years)	Maximum PoF per year [14]
Nozzle Blend Radii	9.67E-7	5.0E-6
Nozzle-to-shell-weld	1.67E-8	5.0E-6

APPENDIX A

LIST OF SUPPORTING FILES

File Name	Description
Blend.INP	VIPERNOZ input file at nozzle blend radii.
Axial_Weld.INP	VIPERNOZ input file at nozzle-to-shell-weld.
Circ_Weld.INP	VIPERNOZ input file at nozzle-to-shell-weld.
Blend.OUT	VIPERNOZ output file at nozzle blend radii.
Axial_Weld.OUT	VIPERNOZ output file at nozzle-to-shell-weld.
Circ_Weld.OUT	VIPERNOZ output file 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