Xcel Energy®

Monticello Nuclear Generating Plant
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L-MT-12-011
10 CFR 50.55a

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

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License No. DPR-22

Supplement to 10CFR50.55a Requests Associated with the Fourth and Fifth Inservice Inspection Ten-Year Intervals

References:

1) BWRVIP-108, "BWR Vessel and Internal Project Technical Basis for the Reduction of Inspection Requirements for Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," dated October 2002.

2) NSPM to NRC, "10 CFR 50.55a Request No. 16: Alternative to Nozzle-to-Vessel Weld and Inner Radius Examinations," dated March 12, 2010. (Accession No. ML100750660)

3) NRC to NSPM, "Monticello Nuclear Generating Plant – Alternative to Nozzle-to-Vessel Weld and Inner Radius Examinations, (TAC No. ME3527)," dated November 24, 2010. (Accession No. ML103190311)

4) NSPM to NRC, "10 CFR 50.55a Requests Associated with the Fifth Inservice Inspection Ten-Year Interval," dated September 28, 2011. (Accession No. ML112720147)

5) Structural Integrity Associates, Inc., Calculation Package, Monticello N2 Nozzle Code Case N-702 Relief Request, File No. 1101463.301

Northern States Power Company, a Minnesota corporation, d/b/a Xcel Energy (hereafter "NSPM"), has identified that Boiling Water Reactor Vessel Internals Project (BWRVIP) report, BWRVIP-108 (Reference 1), contains certain design assumptions that impact its application at the Monticello Nuclear Generating Plant (MNGP). Specifically, BWRVIP-108 assumed a certain number of thermal cycles on the reactor pressure vessel (RPV) nozzles assuming a 40 year design life. Also, the report assumed that the Recirculation Inlet (N2) nozzles accumulated negligible fluence allowing them to be excluded them from the "beltline" region of the RPV. The

BWRVIP-108 report served as part of the basis for two 10 CFR 50.55a requests (References 2 and 4) to implement an alternative, American Society of Mechanical Engineers (ASME) Code Case N-702.

Subsequently, to support use of ASME Code Case N-702, Structural Integrity Associates, Inc. (Reference 5), performed a MNGP site specific analysis for NSPM that evaluates 1) additional thermal cycles assumed during the 60 year design life following approval of license renewal, and 2) additional fluence at the "beltline" region for the RPV N2 nozzles. Enclosure 1 provides this analysis which demonstrates the acceptability of applying ASME Code Case N-702 through the end of the renewed license period of extended operation.

NSPM requests that the U.S. Nuclear Regulatory Commission (NRC) review this site specific analysis together with the pending "10 CFR 50.55a Requests Associated with the Fifth Inservice Inspection Ten-Year Interval," dated September 28, 2011 (Reference 4), specifically as it pertains to 10 CFR 50.55a Request RR-002, "Alternative to the Requirements of Examination Category B-D."

On March 12, 2010, NSPM submitted 10 CFR 50.55a Request No. 16, (Reference 2), pursuant to 10 CFR 50.55a(a)(3)(i) in accordance with ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds." Part of the basis for this 10 CFR 50.55a request was BWRVIP-108. The NRC reviewed this 10 CFR 50.55a request and authorized this alternative for application to MNGP on November 24, 2010, through the end of the Fourth Ten-Year Inservice Inspection (ISI) Interval (Reference 3).

NSPM requests that the NRC review the site specific analysis provided herein, and pursuant to 10 CFR 50.55a(a)(3)(i), reauthorize application of 10 CFR 50.55a Request No. 16 through the Fourth Ten-Year ISI Interval. Reauthorization is necessary to allow use of the ASME Code Case N-702 alternative for the duration of the Fourth Interval, to maintain MNGP compliance with 10 CFR 50.55a(a)(3)(i). NSPM thereby requests that the NRC complete reauthorization prior to the end of the MNGP Fourth Ten-Year ISI Interval, scheduled for August 31, 2012.

Should you have questions regarding this letter, please contact Mr. Randy Rippy at (612) 330-6911.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Resident Inspector, Monticello, USNRC
Project Manager, Monticello, USNRC
BWRVIP Project Manager, USNRC

ENCLOSURE 1

STRUCTURAL INTEGRITY ASSOCIATES, INC.
CALCULATION PACKAGE

MONTICELLO N2 NOZZLE CODE CASE N-702 RELIEF REQUEST

FILE NO.: 1101463.301

12 pages follow

Structural Integrity Associates, Inc.®

CALCULATION PACKAGE

File No.: 1101463.301

Project No.: 1101463

Quality Program: ☒ Nuclear ☐ Commercial

PROJECT NAME:
Monticello N2 Nozzle Code Case N-702 Relief Request

CONTRACT NO.:
00001005 Release 0032

CLIENT:
XCEL Energy

PLANT:
Monticello Nuclear Generating Plant

CALCULATION TITLE:
Evaluation of effect of inspection on the probability of failure for BWR Recirculation Inlet (N2) nozzle-to-shell-welds and nozzle blend radii region at Monticello

Document Revision	Affected Pages	Revision Description	Project Manager Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1 - 10 A-1 - A-2	Initial Issue	Jim Wu	Jim Wu JW 2/2/12 Stan Tang SST 2/2/12

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1.0 OBJECTIVE

This evaluation is to justify the reduction of in-service inspection of the nozzle-to-shell-weld and the nozzle blend radii in the recirculation nozzle (N2) at Monticello Nuclear Generating Plant per code case N-702 for the extended period of operation. The N-702 code case, with appropriate technical justification, may be used as an alternative to the requirements of ASME Section XI, Examination Category B-D.

2.0 METHODOLOGY

The approach was based on the methodology presented in Reference 1. A Monte Carlo simulation was performed using a variance of the program, VIPER [2] with some modifications as described in the following sections. The VIPER program was developed as part of the program contained in Reference 1 for the 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/VIPERNZ program is documented in References 1 and 11.

3.0 DESIGN INPUT

This analysis is intended for evaluating the reduction of inspection based on the probability of failure in the nozzle-to-shell-weld and nozzle blend radius in N2 nozzles at Monticello Nuclear Generating Plant. Some of the input (e.g., pressure through-wall stress distribution, thermal through-wall stress distribution, and weld residual stress through-wall stress distribution) is based on the prior analyses on BWR fleet per References 3 and 4. Others were from Monticello plant specific described below.

Vessel Wall Thickness	= 5.0625" [5]
Vessel Radius	= 103.188" [5]
Vessel Clad Thickness	= 0.125" [5]
Vessel Operating Temperature	= 549°F [10, Page 91]
Operating Pressure	= 1025 psig [10, Page 91]
Radius to Nozzle-to-shell Weld	= 18.25" [6, Figure 1], [17]
End of Life Fluence (54 EFPY/60 years) for N2 nozzles	= 1.01E18 n/cm^2 [7]

The weld chemistries are presented in Table 1.

For the nozzle blend radius region, since the nozzle is a forging, the number of fabrication flaws was assumed to be 0.1 flaws per nozzle. In the weld between the vessel shell and the nozzle, the number of fabrication flaws was assumed to be 1 per nozzle-to-shell-weld. For both locations, the number of stress corrosion initiated flaws was assumed to be 3 per nozzle or per weld.

All random variables were summarized in Table 2 of Reference 8. Most of the input is obtained from Reference 1, except standard deviation for %Cu and %Ni for nozzle blend radii. They are 0.0447 and 0.068 for %Cu and %Ni, respectively and were obtained from BWRVIP-173 [18].

4.0 ASSUMPTIONS

The following assumptions are used in the evaluation [8]:

(1) Fabrication flaw is assumed only due to the weld process in the nozzle-to-shell-weld
(2) One stress corrosion initiated flaw and 0.1 fabrication flaw per nozzle blend radius
(3) One fabrication flaw and one stress corrosion initiated flaw per nozzle/shell weld
(4) Flaw size distribution, PVRUF, is assumed.
(5) Residual stress at the nozzle/shell weld is assumed cosine distribution through the wall thickness with a mean of 8 ksi at surface.
(6) The standard deviation for surface residual stress is assumed to be 5 ksi.
(7) Average upper shelf fracture toughness is 200 ksi$\sqrt{in}$ with a standard deviation of 5 ksi$\sqrt{in}$

5.0 SOFTWARE MODIFICATIONS

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

(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) Preservice 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 1.

6.0 FATIGUE CRACK GROWTH

The fatigue data for A533-B-1 and A508-2 in reactor water environment are reported in Reference 12 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 was chosen for conservatism. The curve fit results of the mean fatigue crack growth law is presented with the Paris Law shown as follows:

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

where a = crack depth
n = cycle
$\Delta K = K_{max} - K_{min}$

A comparison to the ASME Section XI [4] fatigue crack growth law in reactor water environment was done in Reference 8, it shows a very reasonable comparison where Section XI 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:

$$y = -0.3712 + 4.15x \tag{2}$$

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

7.0 STRESS RESULTS AND FATIGUE CYCLE LOADINGS

The stress analyses for the nozzle/shell weld and the nozzle blend radius for the N2 nozzles were presented in Reference 6. The stress analyses were performed for the load cases of unit pressure, and the relevant thermal transients for the N2 nozzles. The through-wall sections were selected based on the thermal transient results. The azimuth locations evaluated were 0°, 90°, 180° and 270° of the nozzles. Two through-wall sections were selected. Section C is at the location of the weld between the RPV and nozzle. Section D is at the blend radius location of the nozzle.

The load cases analyzed for the N2 nozzles include:

(1) Unit pressure
(2) Unit axial load
(3) Unit in-plane moment
(4) Unit out-of-plane moment
(5) Thermal transients depending on the nozzles as described in the following sections

For the selected sections in the N2 nozzles (nozzle blend radius and nozzle-to-vessel shell weld), stresses due to the nozzle axial and moment loads are small compared to the pressure and thermal loadings. Therefore, these load cases were not used in the evaluation.

The thermal transients for the recirculation inlet nozzle are the heat up and sudden pump start of cold recirculation loop. The pressure is maintained at 1050 psig for the sudden pump start transient.

For the 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. The maximum stress among the four azimuth locations was used.

In this section, the maximum stress is at the 90° and 270° in the hoop direction for the combination of pressure and thermal stresses.

The thermal cycles for recirculation inlet nozzle are the number of heat/shutdown cycles (288 cycles Reference 16 for an end of operation time of 60 years), and the number of sudden pump start of cold recirculation loop cycles (10 cycles per page 12 of Reference 13 for an end of operation time of 60 years).

8.0 PROBABILISTIC FRACTURE MECHANICS EVALUATION

The probabilistic evaluation was performed for the case of 25% inspection rate for period of extended operation (assume 70% inspection rate for initial 40 years of operation at the nozzle blend radii and actual inspection rate for initial 40 years of operation at the nozzle-to-shell weld location per Reference 19) and 90% inspection coverage, at 10 years interval for 60 years, for the N2 nozzles.

For the nozzle blend radius region, a nozzle blend radius crack model, [14] 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. Therefore, the maximum stress among the four azimuth locations (0°, 90°, 180° and 270°) was selected.

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 1, 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. It is also shown in the through-wall stress plots in Reference 3 that the difference between the thermal hoop stress and the thermal axial stress is not as much compared to the difference between the pressure hoop stress and the pressure axial stress. Therefore, the probabilistic fracture mechanics evaluation for the nozzle and vessel shell weld would concentrate on the axial crack.

For the nozzle-to-vessel shell weld, the following crack model was used in the evaluation:

(a) Axial elliptical crack model with a crack aspect ratio of $a/l = 0.2$

The inspection PoD curve is the user input of Figure 42 of Reference 8, with an inspection interval every 10 years.

The crack size distribution, PVRUF, is shown in Figure 43 of Reference 8.

The calculation of stress intensity factor is at the deepest point of the crack.

The probability of failure was obtained due to a low temperature over pressurization (LTOP) event at 88 °F and 1150 psi, [15]. The probability of the LTOP event is 1×10^{-3} per year [15].

The analyses were performed using VIPERNOZ, a superset of the program VIPER, [2], with the modifications as described in Section 5.

The number of simulations was 1 million.

9.0 RESULTS OF ANALYSES/CONCLUSIONS

Safety evaluation of proprietary EPRI report, dated December 19, 2007 states that performing the PFM analysis only for recirculation inlet nozzle (N2) is acceptable it has been demonstrated that the recirculation inlet nozzle is limiting for all sensitivity cases. This conclusion is applicable to both nozzle-to-shell weld and nozzle blend radii. In addition, increased inside surface fluence on reactor vessel components results in decreases of fracture toughness, increases of reference temperature and increased susceptibility to SCC and LTOP failures. Based on the test data from the parametric studies in BWRVIP-05 [1], increased fluence results in probabilities of failure orders of magnitude higher than unirradiated cases with similar design parameters in all percentage ISI exams. The N2 nozzles are the only components applicable to the N-702 code case that have accumulated significant fluence, and the thermal transients introduced to the N2 nozzle are as, or more severe, than the transients experienced by the other applicable nozzles. Based on the limiting fluence and stress cases of the N2 nozzle, the results from this analysis shall bound all MNGP nozzle penetrations to the Reactor Pressure Vessel

9.1 Nozzle Blend Radii

The reliability evaluation is presented for the two cases of in-service inspection. The probabilities of failure (PoF) are summarized in Table 2. For the first case, 90% inspection coverage over the 60 years of operation and the second case, 25% inspection rate for period of extended operation of 20 years (assume 70% inspection rate for initial 40 years of operation) at nozzle blend radius. The difference between the total conditional failure probabilities for the two cases is less than 1×10^{-6} per year due to LTOP event. Therefore this analysis demonstrates acceptability of reduced in-service inspection per code case N-702 at the nozzle blend radii in the recirculation nozzle (N2) at Monticello Nuclear Generating Plant for the extended period of operation.

9.2 Nozzle-to-Shell Weld

The reliability evaluation is presented for the two cases of in-service inspection. The probabilities of failure (PoF) are summarized in Table 2. For the first case, 90% inspection coverage over the 60 years of operation and the second case, 25% inspection rate for period of extended operation of 20 years (47% inspection rate for the first 30 years of operation, 78% inspection rate for the last 10 years of operation for initial 40 years of operation per Reference 19) at nozzle-to-vessel shell weld. The difference between the total conditional failure probabilities for the two cases is less than 1×10^{-6} per year due to LTOP event. Therefore this analysis demonstrates acceptability of reduced in-service inspection per code case N-702 at the nozzle-to-shell-weld in the recirculation nozzle (N2) at Monticello Nuclear Generating Plant for the extended period of operation.

10.0 REFERENCES

1. BWRVIP Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Electric Power Research Institute TR-105697, September 1995.
2. VIPER, Vessel Inspection Program Evaluation for Reliability, Version 1.2 (1/5/98), Structural Integrity Associates.
3. SI Calculation W-EPRI-180-301, "RPV Nozzle Stress Analyses," Revision 0.
4. SI Calculation EPRI-180-303, "Deterministic Crack Growth Calculation for BWR Nozzle-to-Shell-welds and nozzle blend radii region," Revision 0.
5. Document NX8290-13, "General Plan Shows Vessel ID (17-2 or 206 Inches) and Vessel Wall Thickness (5-1/5") and Cladding (1/8")," SI File Number 1101463.203.
6. SI Calculation 1000720.301, "Finite Element Stress Analysis of Monticello RPV Recirculation Inlet Nozzle," Revison 0.
7. DIT 19181-01 "MNGP Recirculation Inlet Nozzle-to-shell-welds VIPER Analysis," QF-0545 (FP-E-MOD-11) Revision 3, SI File 1101463.201.

8. 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.\
9. Monticello ART Design Input, SI File Number 1000720.204.
10. Document DBD-B.1.1, "Design Bases Document for Reactor and Vessel Assembly DBD B.1.1," Revision C, SI File Number 1101463.203.
11. BWRVIP Report, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," 1003557, October 2002, SI File Number BWRVIP-01-308.
12. 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
13. Document B.01.01-06, Revision 14, "Operations Manual Section: Reactor and Vessel Assembly B.01.01-06 Figures," SI File Number 1101463.204.
14. 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.
15. NRC, Final Safety Evaluation of the BWR Vessel and Internals, Project BWRVIP-05 Report, TAC # M93925, July 1998.
16. DIT 19181-02 "MNGP Recirculation Inlet Nozzle-to-shell-welds VIPER Analysis," QF-0545 (FP-E-MOD-11) Revision 3, SI File Number 1101463.201.
17. CB&I Drawing No. 7, Revision 9, "12"Ø Nozzle MK. N2 A/K 17'-2" I.D. x 63'-2" Ins. Heads Nuclear Reactor," Monticello Document No. NX-8920-90, SI File No. 1000720.201.
18. BWRVIP Report, "Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials," 1014995, May 2007, SI File Number BWRVIP-01-373.
19. DIT 19181-03 "MNGP Recirculation Inlet Nozzle to Shell Welds VIPER Analysis," QF-0545 (FP-E-MOD-11) Revision 3, SI File Number 1101463.206.

Table 1: Monticello Weld Chemistry

BWR Plant	Inner Dia (in)	Shell Thickness/Path Length (In)	%Cu	%Ni	Initial RTndt(F)
Monticello N2 Nozzle-to-shell-weld	206.4	5.0625	0.1	0.99	-65 [8]
Monticello N2 Blend Radii	206.4	9.4845	0.18	0.86	40 [9]

Note: %Cu and %Ni were obtained from Reference 7. Initial RTndt were obtained from References 8 and 9 Nozzle-to-shell-weld and Blend radii respectively.

Table 2: Probability of Failure Results Summary

	Conditional PoF for 25% In-Service Inspection for period of Extended Operation	Conditional PoF for 90% In-Service Inspection for 60 Years of Operation	Difference in PoF due to LTOP Events between 25% Inspection Rate and 90% Inspection Rate over 60 Years of Operation
Nozzle Blend Radii	5.95E-2	9.24E-3	8.38E-7
Nozzle-to-shell-weld	3.75E-3	1.08E-3	4.45E-8

APPENDIX A

COMPUTER FILES

File Name	Description
RID2I25.INP	VIPERNZ input file for 25% inspection coverage at nozzle blend radii.
RID2I90.INP	VIPERNZ input file for 90% inspection coverage at nozzle blend radii.
RIC2I25.INP	VIPERNZ input file for 25% inspection coverage at nozzle-to-shell-weld.
RIC2I90.INP	VIPERNZ input file for 90% inspection coverage at nozzle-to-shell-weld.
RID2I25.OUT	VIPERNZ output file for 25% inspection coverage at nozzle blend radii.
RID2I90.OUT	VIPERNZ output file for 90% inspection coverage at nozzle blend radii.
RIC2I25.OUT	VIPERNZ output file for 25% inspection coverage at nozzle-to-shell-weld.
RIC2I90.OUT	VIPERNZ output file for 90% inspection coverage at nozzle-to-shell-weld.
VIPERNOZ1P3.EXE	VIPERNZ executable program