Xcel Energy

November 12, 2008

L-PI-08-097
10 CFR 54

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Responses to NRC Requests for Additional Information Dated November 4, 2008 Regarding Application for Renewed Operating Licenses

By letter dated April 11, 2008, Northern States Power Company, a Minnesota corporation, (NSPM) submitted an Application for Renewed Operating Licenses (LRA) for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. In a letter dated November 4, 2008, the NRC transmitted Requests for Additional Information (RAIs) regarding that application. This letter provides responses to those RAIs.

Enclosure 1 provides the text of each RAI followed by the NSPM response.

If there are any questions or if additional information is needed, please contact Mr. Eugene Eckholt, License Renewal Project Manager.

Summary of Commitments

This letter contains no new commitments or changes to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on November 12, 2008.

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Northern States Power Company - Minnesota



A133
NRR

Enclosure (1)

cc:

Administrator, Region III, USNRC
License Renewal Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Prairie Island Indian Community ATTN: Phil Mahowald
Minnesota Department of Commerce

## Enclosure 1
## Responses to NRC Requests for Additional Information Dated November 4, 2008

### RAI B.2.1.34

(A) Section 6.0 in aging management program XI.31, "Reactor Vessel Surveillance," in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Volume 2, Revision 1 states that if an applicant has a surveillance program that consists of capsules with a projected fluence of less than the projected 60 year fluence at the end of 40 years, at least one capsule is to remain in the reactor vessel and tested during the extended period of operation.

Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2 have tested four of the six surveillance capsules in each unit to date, and the latest capsules of Units 1 and 2 were tested at projected fluence values which are less than 60 year fluence. To ensure that the applicant is in compliance with the aforementioned GALL Report requirement, one of the two remaining capsules in each unit should be tested during the extended period of operation and the following relevant information regarding the testing of the capsules should be provided to the staff.

(1) Applicant's plan to test an additional surveillance capsule from each unit
(2) The projected refueling outages of withdrawal for each unit
(3) Projected neutron fluence value for each capsule at the time of withdrawal

(B) The staff requests that the applicant confirm that the withdrawal schedule of the capsules to be used for future tests during the extended period of operation is consistent with the requirements, specifically the limitations on lead factor, specified in paragraph 7.6.2 of the American Society of Testing Materials (ASTM) E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

(C) The staff requests that the applicant confirm that untested surveillance capsules (standby capsules) will be stored for future use at PINGP, Units 1 and 2.

### NSPM Response to RAI B.2.1.34

Part (A)

(A)(1) Initially, six (6) surveillance capsules were installed in each Unit at PINGP. Four capsules were removed per an established withdrawal schedule. The last capsule to be removed from Unit 1 was Capsule 'S' at 18.12 EFPY and the last capsule removed from Unit 2 was Capsule 'P' at 17.24 EFPY. Two spare capsules remain installed in each reactor vessel. To account for license renewal, one of the two remaining capsules from each unit will be withdrawn after the capsule has received a neutron fluence equivalent to the 60-year fluence. After capsule removal, the surveillance specimens will be tested in accordance with the requirements of 10 CFR 50, Appendix H and ASTM E 185-82. The results of materials testing, fluence analysis, and effective full power years (EFPY) from startup are used to predict the effects of neutron embrittlement through the end of extended life. The remaining spare capsules will stay in the reactor vessel to provide meaningful metallurgical data for potential future license renewals.

(A)(2) Prairie Island Unit 1 surveillance capsule withdrawal is planned for refueling outage 1R27, expected to occur in 2011. Prairie Island Unit 2 surveillance capsule withdrawal is planned for refueling outage 2R27, expected to occur in 2012.

(A)(3) The maximum reactor vessel fluence at 54 EFPY for Unit 1 is projected to be 5.162E19 n/cm$^2$ (E>1.0 MeV), and for Unit 2 is projected to be 5.196E19 n/cm$^2$ (E>1.0 MeV). Per the requirements of ASTM E 185-82, Section 7.6, the surveillance capsules should be removed when their neutron fluence exceeds the new peak End-Of-Life (EOL) vessel fluence (i.e.; 5.162E19 n/cm$^2$ for Unit 1 at 54 EFPY, and 5.196E19 n/cm$^2$ for Unit 2 at 54 EFPY), but prior to exceeding twice that fluence exposure (i.e.; 1.032E20 n/cm$^2$ for Unit 1, and 1.039E20 n/cm$^2$ for Unit 2). Calculations have been performed to determine the earliest withdrawal times for the remaining capsules. The results are presented below:

<u>Unit 1</u>
Capsule T (Lead Factor = 1.89): Removal Time ≥ 24.2 EFPY
Capsule N (Lead Factor = 1.77): Removal Time ≥ 26.5 EFPY

<u>Unit 2</u>
Capsule N (Lead Factor = 1.72): Removal Time ≥ 28.5 EFPY
Capsule S (Lead Factor = 1.72): Removal Time ≥ 28.5 EFPY

As of July 1, 2008, the total lifetime performance of both Units is 29.0 EFPY.

Additional calculations have been performed to estimate the neutron fluence of each capsule at the time of next withdrawal. Current plans are to remove one capsule from each Unit during refueling outages 1R27 and 2R27. The following are the projected fluence values for each capsule at that time:

<u>Unit 1 (Next capsule removal planned for 1R27, Withdrawal EFPY = 31.6)</u>
Capsule T (Lead Factor = 1.89): Projected Neutron Fluence = 6.292E19 n/cm$^2$
Capsule N (Lead Factor = 1.77): Projected Neutron Fluence = 5.893E19 n/cm$^2$

<u>Unit 2 (Next capsule removal planned for 2R27, Withdrawal EFPY = 32.2)</u>
Capsule N (Lead Factor = 1.72): Projected Neutron Fluence = 5.739E19 n/cm$^2$
Capsule S (Lead Factor = 1.72): Projected Neutron Fluence = 5.739E19 n/cm$^2$

Part (B)

(B) The surveillance capsule withdrawal schedule is generated based upon the requirements specified in ASTM E 185-82, Section 7.6. Six surveillance capsules were installed in each Unit at PINGP. Four capsules have been removed per an established withdrawal schedule. The last capsule to be removed from Unit 1 was Capsule 'S' at 18.12 EFPY and the last capsule removed from Unit 2 was Capsule 'P' at 17.24 EFPY. In accordance with the requirements of ASTM E 185-82, Section 7.6.2, one of the remaining capsules will be withdrawn from each Unit when its neutron fluence exposure exceeds the new peak EOL (54 EFPY) vessel fluence, but prior to exceeding twice that fluence exposure. A revised capsule withdrawal schedule will be submitted for NRC approval prior to implementation in accordance with the requirements of 10 CFR 50, Appendix H.

Part (C)

(C) To account for license renewal, one of the two remaining capsules from each Unit will be withdrawn after the capsule receives a neutron fluence equivalent to the 60-year vessel fluence. The remaining spare capsules (one per Unit) will stay in the reactor vessel to provide meaningful metallurgical data for potential future license renewals. As discussed in LRA Section B2.1.34 (Page B-69) and Commitment No. 27, the PINGP Reactor Vessel Surveillance Program will be enhanced by adding a requirement to the program to ensure that in the event spare (standby) capsules are withdrawn, the untested capsules are placed in storage and maintained for future insertion at PINGP Units 1 and 2.

## RAI 4.2.2

Based on the beltline material "Circumferential Weld – Nozzle Shell Forging B to Intermediate Shell Forging C", the staff assumes that there is a small line nozzle penetration in the beltline region of the RPV. The staff requests the applicant to confirm this detail.

## NSPM Response to RAI 4.2.2

The "Circumferential Weld - Nozzle Shell Forging B to Intermediate Shell Forging C," is a circumferential weld that connects the Reactor Pressure Vessel (RPV) nozzle (or upper) shell forging B to the RPV intermediate shell forging C. There is no small line nozzle penetration in the beltline region of the PINGP RPV. The RPV beltline region is depicted in PINGP USAR Figure 4.7-4.

## RAI 4.2.3

In Table 4.2-5 of the PINGP license renewal application (LRA), the $RT_{NDT(u)}$ for beltline material "Lower Shell Forging D (22642)" is listed as -4 °F. However, the value of $RT_{NDT(u)}$ for this beltline material is listed as 2 °F in the PINGP reactor vessel integrity database (RVID). Provide information which documents where the value of -4 °F comes from and demonstrate that is applicable to this forging.

## NSPM Response to RAI 4.2.3

The initial $RT_{NDT(u)}$ of -4 °F for PINGP Unit 2, lower shell forging D (22642) is consistent with the $RT_{NDT(u)}$ value used in the current "PINGP Units One and Two Pressure and Temperature Limits Report", Revision 3 (Effective until 35 EFPY) and reported in WCAP-14637, "Prairie Island Unit 2 Heatup and Cooldown Limit Curves for Normal Operation", Revision 3, December 1999 (ML023230354 and ML003703560, respectively).

The initial $RT_{NDT(u)}$ of -4 °F for PINGP Unit 2, lower shell forging D (22642) is obtained from the PINGP USAR, Table 4.7-9, which references, as a data source, Structural Integrity Report SIR-99-075, Revision 2, "Update of the Response to Generic Letter 92-01 Reactor Vessel Structural Integrity for Prairie Island Units 1 and 2", November 1999 (ML993410414). This report was previously transmitted from NSP to the NRC via letter entitled "Comprehensive Revised Response to Generic Letter 92-01", dated November 10, 1999 (ML993330371). The initial $RT_{NDT(u)}$ of -4 °F for lower shell forging D (22642) is documented as a change on page viii, Table 2-4, and Table 2-5 of SIR-99-075, Revision 2. Section 7.0 of SIR-99-075 provided a summary of the PINGP information in the RVID2 database, and proposed a number of revisions to the database. As noted in Table 7-3 (page 7-4) of SIR-99-075, the proposed changes included a correction in $RT_{NDT(u)}$ from 2 °F to -4 °F for lower shell forging D (22642). Therefore, the -4 °F value of $RT_{NDT(u)}$ is applicable to lower shell forging D.

## RAI: 4.2.4

Per the requirements of Title 10 of the Code of Federal Regulations, Section 54.21(c)(1)(i) and the staff guidance in NUREG 1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," the projected neutron fluence and subsequent adjusted reference temperature (ART) values at the end of the extended period of operation are reviewed to verify that they are bound by the embrittlement assumed in the existing P-T limit analysis. Therefore, the staff requests the applicant to provide current ¼ T and ¾ T ART values for all the beltline materials.

## NSPM Response to RAI 4.2.4

As discussed in Section 4.2.4 (Page 4.2-9) of the PINGP LRA, PINGP has selected 10 CFR 54.21(c)(1)(iii) and not (i) to manage the TLAA associated with P-T limits. Updated P-T limits for the period of extended operation will be managed by the Reactor Vessel Surveillance Program. As stated in Appendix B2.1.34 (Page B-68) of the PINGP LRA, "The Reactor Vessel Surveillance Program manages updates of pressure-temperature operating limitations and the surveillance specimen withdrawal schedule, as needed, consistent with plant Technical Specifications, the Pressure and Temperature Limits Report, and 10 CFR 50.60 and 10 CFR 50, Appendix H."

In response to the NRC's request, ¼T and ¾T adjusted reference temperature (ART) values are provided below for all the beltline materials of PINGP Unit 1 and Unit 2. ART values are provided at 35 EFPY, which is the basis for the current P-T limits, as well as at 54 EFPY, which is assumed to define the end of the period of extended operation. The 54 EFPY ART values were calculated using the procedures given in Regulatory Guide 1.99, Revision 2, and are provided for information only.

**PINGP Unit 1 ART Values**

| Unit 1-Beltline ID | Material Type | ART (°F) 1/4T at 35 EFPY | ART (°F) 3/4T at 35 EFPY | ART (°F) 1/4T at 54 EFPY | ART (°F) 3/4T at 54 EFPY |
|---|---|---|---|---|---|
| Nozzle Shell Forging B-21744/38384 | A 508-3 | 87[1] | 75[1] | 83[1] | 72[1] |
| Lower Shell Forging D-21887/38530 | A 508-3 | 85[1] | 76[1] | 88[1] | 79[1] |
| Int. Shell Forging C-21918/38566 | A 508-3 | 103[1]<br>117[2] | 94[1]<br>105[2] | 106[1]<br>120[2] | 97[1]<br>109[2] |
| Circ Weld-Int. Shell Forging C to Lower Shell Forging D, 1752 | UM40, Flux UM89 | 131[1]<br>145[2] | 116[1]<br>128[2] | 135[1]<br>149[2] | 120[1]<br>133[2] |
| Circ Weld-Nozzle Shell Forging B to Int. Shell Forging C, 2269 | UM40, Flux UM89 | 154[1] | 136[1] | 149[1] | 131[1] |

1-Position (1.1) - No Surveillance Data
2-Position (2.1) - Surveillance Data

**PINGP Unit 2 ART Values**

| Unit 2-Beltline ID | Material Type | ART (°F) 1/4T at 35 EFPY | ART (°F) 3/4T at 35 EFPY | ART (°F) 1/4T at 54 EFPY | ART (°F) 3/4T at 54 EFPY |
|---|---|---|---|---|---|
| Nozzle Shell Forging B-22231/39088 | A 508-3 | 71[1] | 61[1] | 67[1] | 57[1] |
| Lower Shell Forging D-22642 | A 508-3 | 95[1]<br>106[2] | 84[1]<br>94[2] | 97[1]<br>109[2] | 87[1]<br>97[2] |
| Int. Shell Forging C-22829 | A 508-3 | 104[1] | 95[1] | 106[1] | 97[1] |
| Circ Weld-Int. Shell Forging C to Lower Shell Forging D, 2721 | UM40, Flux UM89 | 91[1]<br>99[2] | 79[1]<br>82[2] | 93[1]<br>103[2] | 83[1]<br>86[2] |
| Circ Weld-Nozzle Shell Forging B to Int. Shell Forging C, 1752 | UM40, Flux UM89 | 122[1]<br>134[2] | 106[1]<br>116[2] | 116[1]<br>127[2] | 100[1]<br>109[2] |

1-Position (1.1) - No Surveillance Data
2-Position (2.1) - Surveillance Data

The limiting Unit 1 material at 35 EFPY is the Circumferential Weld - Nozzle Shell Forging B to Intermediate Shell Forging C (2269) with a 1/4T ART of 154 °F and 3/4T ART of 136 °F. The limiting Unit 2 material at 35 EFPY is the Circumferential Weld - Nozzle Shell Forging B to Intermediate Shell Forging C (1752) with a 1/4T ART of 134 °F and 3/4T ART of 116 °F. The 54 EFPY adjusted reference temperatures (1/4T and 3/4T) of these limiting materials have decreased relative to these 35 EFPY ART values. Based on a comparison of the 54 EFPY ARTs and 35 EFPY ARTs for the current limiting materials, it is anticipated that PINGP will have sufficient operating margin to conduct plant heatups and cooldowns for the period of extended operation.

## RAI 4.7.2

In Section 4.7.2 of the PINGP LRA, the applicant states that the number of design cycles and transients assumed in topical report WCAP-15338, "A Review of Cracking with Weld Deposited Cladding in Operating PWR Plants," analyses bound the number of design cycles and transients projected for 60 years of operation as presented in Table 4.3-1 of the PINGP LRA. However, the WCAP-15338 report does not explicitly state the number of design cycles and transients used in the analyses of crack growth for 60 years of operation of the reactor pressure vessel. The staff requests the applicant to provide the bounding number of design cycles and transients that were used in the WCAP-15338 report analyses.

## NSPM Response to RAI 4.7.2

The numbers of design cycles used in the fatigue crack growth evaluation is reported on page 9-10 of WCAP-15338-A. For convenience, the table is reproduced below as it appears in WCAP-15338-A.

| REACTOR COOLANT SYSTEM TRANSIENTS FOR 40 YEARS | |
|---|---|
| Transient Identification | WCAP-15338 |
| | Number for 40 Years* |
| Normal Conditions | |
| 1. Heatup and Cooldown at 100F/hr | 200 each |
| 2. Load Follow Cycles (unit loading and unloading at 5% of full power / min) | 18300 |
| 3. Step load increase and decrease of 10% of full power | 2000 each (20% step load changes) |
| 4. Large step load decrease, with steam dump | 200 |
| 5. Steady state fluctuations, initial / random | 105E5 / 3.0E6 |
| 6. Feedwater Cycling at Hot Shutdown | 2000 |
| 7. Loop Out of Service, shutdown / startup | 80/70 |
| 8. Unit loading and unloading between 0% and 15% of full power | |
| 9. Boron Concentration Equalization | 26400 |
| 10. Refueling | 80 |
| | |
| Upset Conditions | |
| 1. Loss of load, without immediate turbine or reactor trip | 200 |
| 2. Loss of power (blackout with natural circulation in the RCS) | 40 |
| 3. Loss of flow (partial loss of flow, one pump only) | 80 |
| 4. Reactor trip<br>- No cooldown<br>- Cooldown, no safety injection<br>- Cooldown with SI | <br>230<br>160<br>10 |
| 5. Inadvertent RCS depressurization | 20 |
| 6. Inadvertent startup of an inactive loop | 10 |
| 7. Control rod drop | 80 |
| 8. Inadvertent Safety Injection | 60 |
| *Excessive Feedwater Flow* | 30 |

| | |
|---|---|
| Test Conditions | |
| 1. Turbine roll test | 80 |
| 2. Primary side hydrostatic test | 10 |
| 3. Primary Side Leakage Test | 280 primary/secondary |

*The 60-year number of transients is 1.5 times the 40-year number

As indicated by the footnote, the 60-year fatigue crack growth analysis assumed a 50% increase in the number of design transients compared to the standard 40-year set. The numbers of design cycles and transients assumed in the WCAP-15338 analysis bound the numbers of design cycles and transients projected for 60 years of PINGP operation, as reported in Table 4.3-1 of the LRA.