UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 26, 2023

Mr. Thomas A. Conboy
Site Vice President
Northern States Power Company – Minnesota
Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1 – REVIEW OF REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM CAPSULE N TECHNICAL REPORT (EPID L-2022-LRO-0160)

Dear Mr. Conboy:

By letter dated March 7, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22067A147), Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (NSPM, the licensee) submitted report WCAP-18660-NP, "Analysis of Capsule N from the Xcel Energy Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program." The report was provided in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) part 50, appendix H, section IV. Testing was performed in accordance with American Society for Testing and Materials Standard (ASTM) E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," as specified in 10 CFR part 50, appendix H, paragraph III.B.1.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of NSPM's submittal as documented in the enclosed staff evaluation. The NRC staff concludes that NSPM has provided the information required by the regulations and that no additional follow-up is required at this time. This completes the NRC staff's efforts for EPID L-2022-LRO-0160.

If you have any questions, please contact me at 301-415-0680 or via email at Brent.Ballard@nrc.gov.

Sincerely,

/RA/

Brent T. Ballard, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-282

Enclosure:
Review of Reactor Vessel
Material Surveillance Program
Capsule N Technical Report

cc: Listserv

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NUCLEAR REGULATORY COMMISSION
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OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW OF REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

CAPSULE N TECHNICAL REPORT

NORTHERN STATES POWER COMPANY

PRIARIE ISLAND NUCLEAR GENERATING PLANT UNIT 1

DOCKET NO. 50-282

1.0 INTRODUCTION

By letter dated March 7, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22067A147), Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (NSPM, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) an evaluation of the testing results of reactor vessel radiation surveillance program Capsule N for the Prairie Island Nuclear Generating Plant (Prairie Island), Unit 1, in accordance with the Title 10 of the *Code of Federal Regulations* (10 CFR), part 50, appendix H, “Reactor Vessel Material Surveillance Program Requirements.” The evaluation report is titled WCAP-18660-NP, Revision 0, “Analysis of Capsule N from the Xcel Energy Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program,” dated November 2021.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR, part 50, appendix H, requires licensees to implement a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light-water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment.

Paragraph IV.A of appendix H to 10 CFR, part 50, specifies that a summary technical report for each capsule withdrawal and the associated test results must be submitted within 18 months of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

Paragraph IV.B of appendix H to 10 CFR, part 50, requires that capsule evaluation reports include all data specified by American Society for Testing and Materials (ASTM) Standard Practice E 185-82 and the results of all fracture toughness tests conducted on the surveillance capsule materials in both the unirradiated and irradiated condition.

Paragraph IV.C of appendix H to 10 CFR, part 50, requires that if a change in the technical specifications (TSs) is required, either in the pressure-temperature (P/T) limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised TSs must be provided with the report.

The NRC Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ML003740284), provides guidance on general procedures acceptable to the NRC staff for calculating effects of neutron radiation embrittlement of low-alloy steels used for light-water-cooled reactor vessels.

3.0 NRC STAFF EVALUATION

3.1 Surveillance Capsule Program

Irradiation surveillance of the reactor vessel is necessary to assure that the vessel material will maintain its fracture toughness throughout the service life of the plant. The surveillance capsule contains both dosimeters as well as archival material samples to be irradiated to levels comparable to those expected to be accrued by the reactor vessel at the end of its licensed period. Under the program, fracture toughness test data is obtained from the material specimens exposed in the surveillance capsules which are withdrawn periodically from the reactor vessel.

The Prairie Island, Unit 1, reactor pressure vessel (RPV) material surveillance program was developed based on ASTM E 185-70 "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The program contains a total of six capsules. Capsule N is the fifth capsule that has been removed. The licensee removed Capsule N from the reactor vessel at 40.06 effective full-power years (EFPY).

3.2 Neutron Fluence Evaluation

A fluence evaluation utilizing the neutron transport and dosimetry cross-section libraries was derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). The licensee reported that Capsule N was removed at 40.06 EFPY and received a fluence of 8.45×10^{19} n/cm^2 (E >1.0 MeV).

In WCAP-18660-NP, Revision 0, the licensee stated that the transport calculations supporting the analysis of the fluence of Capsule N were carried out using the RAPTOR-M3G computer code. While RAPTOR-M3G has been reviewed and approved by the NRC staff for referencing in licensing applications via WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" (ML18204A010), the licensee has not been approved by the NRC to incorporate this methodology into the licensing basis for Prairie Island, Unit 1. From the NRC staff's review of appendix A of the licensee's submittal, which compares the measured capsule dosimetry results with those of previously withdrawn capsules, the NRC finds that no safety significant issue is presented or implied by the results of the capsule surveillance report. This statement should not be construed as NRC approval for use of RAPTOR-M3G as a method for evaluation for Prairie Island, Unit 1. The use of RAPTOR-M3G would require appropriate incorporation of the methodology into the Prairie Island, Unit 1, licensing basis.

The licensee projected the peak clad/base metal interface vessel fluence at 54 EFPY (end-of-license extension) of plant operation to be 5.63×10^{19} n/cm^2 (E > 1.0 MeV). Using the RG 1.99, Revision 2, attenuation formula and a vessel thickness of 6.692 inches, the licensee calculated vessel peak quarter-thickness (1/4T) and three-quarter thickness (3/4T) fluence of 3.77×10^{19} n/cm^2 and 1.69×10^{19} n/cm^2, respectively.

3.3 Material Test Results

The licensee performed mechanical tests of the Charpy V-notch and tensile specimens in Capsule N in accordance with 10 CFR, part 50, appendix H, and ASTM Specification E 185-82.

3.3.1 Transition Temperature Shift

The reactor vessel intermediate shell forging C (Heat # 21918) Charpy specimens, oriented with the longitudinal axis of the specimen parallel to the major working direction (tangential orientation), resulted in an irradiated 30 ft-lb (foot-pound) transition temperature of 126.5 °F (degree Fahrenheit). This results in a 30 ft-lb transition temperature increase of 155.6 °F.

The reactor vessel intermediate shell forging C (Heat # 21918) Charpy specimens, oriented with the longitudinal axis of the specimen perpendicular to the major working direction (axial orientation), resulted in an irradiated 30 ft-lb transition temperature of 116.5 °F. This results in a 30 ft-lb transition temperature increase of 147.3 °F.

The weld material (Heat # 1752, Flux Type UM89, Lot # 1230) Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of 154.0°F. This results in a 30 ft-lb transition temperature increase of 219.0°F.

The reactor vessel heat-affected zone (HAZ) material Charpy specimens resulted in an irradiated 30 ft-lb transition temperature of -54.9 °F. This results in a 30 ft-lb transition temperature increase of 210.1 °F.

The licensee’s test results show that the measured shifts in the 30 ft-lb transition temperature of all the surveillance materials are higher than the RG 1.99, Revision 2, position 1.1, predictions. The test results show that the RG 1.99, Revision 2 prediction is not accurate and not conservative when the RPV material is irradiated to a high neutron fluence. The NRC staff will consider the implications of the non-conservativism in future licensing actions, such as when the licensee evaluates RPV embrittlement for license renewal.

3.3.2 Upper Shelf Energy

The average upper-shelf energy of intermediate shell forging C (Heat # 21918) (tangential orientation) resulted in an average energy decrease of 20.5 ft-lb. This decrease results in an irradiated average upper-shelf energy of 138 ft-lb for the tangentially oriented specimens.

The average upper-shelf energy of intermediate shell forging C (Heat # 21918) (axial orientation) resulted in an average energy decrease of 23.6 ft-lb. This decrease results in an irradiated average upper-shelf energy of 119.5 ft-lb for the axially oriented specimens.

The average upper-shelf energy of the surveillance program weld material (Heat # 1752) Charpy specimens resulted in an average energy decrease of 3.5 ft-lb. This decrease results in an irradiated average upper-shelf energy of 79.0 ft-lb.

The average upper-shelf energy of the HAZ material Charpy specimens resulted in an average energy decrease of 105.9 ft-lb. This decrease results in an irradiated average upper-shelf energy of 105.2 ft-lb.

The measured percent decreases in upper-shelf energy of all the surveillance materials are less than the RG 1.99, Revision 2, position 1.2, predictions. These results show that RG 1.99 method is conservative in the predicting the upper-shelf energy of the reactor vessel material.

3.3.3 Credibility Evaluation

The NRC staff notes that the credibility assessment performed in appendix D of WCAP-18660-NP, Revision 0, found the surveillance weld and surveillance forging data to be “non-credible” in accordance with the five criteria listed in RG 1.99, Revision 2. The licensee stated in its evaluation that although the surveillance program materials did not meet Criterion 3, the surveillance data may still be used in determining the upper-shelf energy decrease in accordance with RG1.99, Revision 2, position 2.2. The NRC staff reviewed the credibility evaluation and had no objections. However, surveillance data determined to be “non-credible” may still need to be factored into licensing calculations, such as P/T limits.

4.0 CONCLUSION

The licensee is expected to incorporate the updated surveillance data into the next revision of the Prairie Island, Unit 1, P/T limits report, as required by 10 CFR, part 50, appendix G. The licensee specified an expected submittal date in their submission and has submitted a license amendment request (LAR) to revise Prairie Island TS 5.6.6, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” to incorporate use of newer analytical methods to determine the RCS P/T limits and cold overpressure mitigation system setpoints (ML22343A257). The LAR is currently under review by NRC staff.

The NRC staff finds that the licensee’s report of reactor vessel surveillance capsule N from Unit 1 satisfies the requirements of 10 CFR, part 50, appendix H, and that the licensee performed tests and calculations based on 10 CFR 50, appendix H and RG 1.99, Revision 2, appropriately. The staff also concludes that the licensee’s submittal meets the reporting requirements in section IV of 10 CFR, part 50, appendix H.

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Date: September 26, 2023

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1 – REVIEW OF REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM CAPSULE N TECHNICAL REPORT (EPID L-2022-LRO-0160) DATED SEPTEMBER 26, 2023

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