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

50.59 Evaluation Summary Report

With this letter, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), submits two enclosures. Enclosure 1 contains descriptions and summaries of safety evaluations for changes, tests, and experiments made under the provisions of 10 CFR 50.59 during the period since the last update. Enclosure 2 contains a discussion of changes to regulatory commitments made within our Regulatory Commitment Change Process during the period since the last update.

Summary of Commitments

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

Kevin Davison

Kevin Davison
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Northern States Power Company - Minnesota

Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

REPORT OF CHANGES, TESTS, AND EXPERIMENTS - DECEMBER 2013

Below is a brief description and a summary of the safety evaluation for each of those changes, tests, and experiments which were carried out at the Prairie Island Nuclear Generating Plant by Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), without prior Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR 50.59.

50.59 Evaluation No. 1025 R6 - D104.1, Zebra Mussel Treatment

Description of Change

Chemically treat portions of the Circulating Water System (CW), the Cooling Water System (CL) and Fire Protection System (FP) to eradicate zebra mussel population within the Prairie Island facility per D104.1, Zebra Mussel Treatment: Circulating Water System. This evaluation is being revised to incorporate EC 22641 which evaluates the effects of a zebra mussel treatment, per D104.1, on plant equipment with consideration to the largest mussel population. This evaluation also incorporates the use of the general chemical "quat" (non-oxidizing quaternary amine) and specifically for the Fall 2013 campaign, the chemical Nalco H-130M.

While simultaneous treatment of Unit 1/Unit 2 for zebra mussels creates the potential to challenge plant systems, the design of the plant screens, strainers, and support systems, and procedural controls within D104.1 minimize the potential for plugging. The performance of D104.1 SHALL be considered a special test/procedure and as such requires evaluation prior to performance.

Summary of 50.59 Evaluation

The treatment for zebra mussel control has the potential to affect the cooling water, circulating water and fire protection systems. The evaluation has determined that this activity does not result in more than a minimal increase in the frequency of occurrence of an accident or the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR or any pending submittal. It has also been determined that there will be no effect on off-site or on-site dose resulting in more than a minimal increase in the consequences of an accident or malfunction of an SSC important to safety previously evaluated in the USAR or any pending submittal. The evaluation shows that the activity does not create a possibility for an accident of a different type than has already been evaluated in the USAR and pending submittals and that there are no new failure modes that are not already bounded by existing analyses that would result in a possibility for a malfunction of an SSC important to safety with a

different result than previously evaluated. Finally, the activity has been found to not result in a design basis limit for a fission product barrier being exceeded or altered.

50.59 Evaluation No. 1091 R0 - Changes to Lithium Control Strategy in RPIP 3006 "PRIMARY WATER CHEMISTRY GUIDELINES"

Description of Change

The activity being evaluated is the changing of the Reactor Coolant System's (RCS) maximum lithium concentration from 5.0 ppm to 6.0 ppm for a 24 hour period during initial startup of a fuel cycle and an additional 24 hour period during another startup within the first 4,000 megawatt day per metric ton of uranium of each fuel cycle. This change could impact corrosion phenomena of RCS materials including general corrosion, primary water stress corrosion cracking, and fuel cladding oxidation. This change is being made to allow the RCS pH to be maintained at or above 6.9 during reactor startup and power escalation. Maintaining the RCS pH value at or above 6.9 reduces general corrosion and thus reduces radiation levels around the RCS due to fewer activation products and less crud available for deposition on the fuel cladding.

Summary of 50.59 Evaluation

The proposed changes are consistent with the guidelines contained in the Electric Power Research Institute (EPRI) Pressurized Water Reactor Primary Water Chemistry Guidelines. The changes are also consistent with fuel vendor's chemistry control guidelines. Thus, the structural integrity of the RCS boundary and fuel cladding are unaffected by the change. Therefore there is no change to the likelihood of an accident or malfunction of a component and no new accident or malfunction types are created. The effect on tritium production is negligible. In addition, tritium is not included in the radioactive source terms used in the accident dose analyses. Therefore there is no change to the consequences of an accident or malfunction. Since the proposed change is consistent with the fuel vendor's chemistry guidelines, the fuel cladding will continue to meet its design basis limits as a fission product barrier.

50.59 Evaluation No. 1093 R0 - Use of RELAP 5 vs. RELAP 3 for FW Line Break, HELB

Description of Change

This evaluation determines whether Nuclear Regulatory Commission (NRC) approval is required for the use of RELAP5/MOD2-B&W methodology for analyzing a main feedwater line break outside containment. Evaluations for a feedwater line break thrust loads outside containment are discussed in Updated Final Safety Analysis Report (UFSAR), Appendix I. The current licensing basis describes the method of evaluation utilizing RELAP3. The use of RELAP3 is being discontinued in industry due to more recent revisions to the code. The NRC has reviewed and approved the use of

RELAP5/MOD2-B&W in a Safety Evaluation Report (SER) dated March, 1995. This evaluation determines whether the limitations and requirements for the use of RELAP5/MOD2-B&W are applicable to the feedwater line break outside containment at Prairie Island.

Summary of 50.59 Evaluation

This evaluation has determined that NRC approval is not required to change the methodology as described herein, namely the use of RELAP5/MOD2 for modeling Main Feedwater Line High Energy Line Break analysis. The evaluation shows that this change is not a departure for a method of evaluation described in the UFSAR since it involves the use of a new NRC-approved methodology (e.g., new or upgraded computer code) to reduce uncertainty, provide more precise results or other reason and has been shown to be (a) based on sound engineering practice, (b) appropriate for the intended application and (c) within the limitations of the applicable Safety Evaluation Report.

Therefore, this change does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

50.59 Evaluation No. 1094 R0 - Unit 2 Cycle 27 Core Reload Modification

Description of Change

This activity will replace depleted fuel from the Unit 2 Cycle 26 reactor core with 45 feed (fresh) fuel assemblies and 8 reinsert fuel assemblies from the spent fuel pool last used in Unit 2 Cycle 25. This will allow the Unit 2 reactor to produce power at its rated capacity in Cycle 27 for approximately 18 months. This activity is required because the fuel in the cycle 26 core is depleted to a state that no longer allows for full power operation. This evaluation is valid for operation of Unit 2 Cycle 27 in Modes 1 through 6.

Summary of 50.59 Evaluation

The UFSAR Chapter 14 evaluations performed by Westinghouse demonstrate that the Prairie Island Unit 2 Cycle 27 reload design and associated COLR do not result in the licensed safety limits for any accident being exceeded. The Cycle 27 design is consistent with the description of the core in the USAR, including changes associated with License Amendment 199/187 which approved use of Optimized ZIRLOTM High Performance Fuel Cladding Material. The core contains 121 fuel assemblies using a 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the UFSAR. The only change from Cycle 26 is the addition of new 422V+ fuel assemblies and the rearrangement of used fuel assemblies of the 422V+ and Optimized Fuel Assembly (OFA) V+ designs. This change results in an isotopic distribution of the core that changes the core physics parameters. The effect of these changes in the cycle

physics parameters on cycle operation and accident analyses have been evaluated using NRC-approved methods.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore, the reload modification for Unit 2 Cycle 27 is consistent with Prairie Island's Current Licensing Basis.

50.59 Evaluation No. 1095 R0 - Revised Steamline Break Containment Response Analysis of Record for Unit 2

Description of Change

The purpose of this evaluation is to implement a new calculation as the Analysis of Record for the Unit 2 containment response to a Main Steam Line Break inside of containment. This activity is required due to performance testing results showing an increase in the Auxiliary Feedwater flow. The increased flow will provide additional mass and energy which is an input to the limiting peak containment pressure and temperature cases for this event.

Summary of 50.59 Evaluation

This activity does not require prior NRC approval as the new analysis of record, including accounting for a larger Auxiliary Feedwater flow, used the Prairie Island current licensing basis NRC approved methodology and the results showed that the design limits as currently described in the Prairie Island Updated Final Safety Analysis Report are met. Thus, there is no increase to the consequences of an accident or malfunction. In addition, this activity does not impact equipment operations, performance and reliability thus there is no change to the frequency of an accident, likelihood of a malfunction, possibility of a new accident, or possibility of a malfunction with a different result.

50.59 Evaluation No. 1096 R0 - Relay Room Integrity Test

Description of Change

The proposed activity is an integrity test of the Relay Room to determine the amount of leakage out of the room. It is not clear that current relay room leakage rate values were determined through appropriate testing. Since the leakage rate identified for the relay and computer room cannot be verified an integrity test must be performed.

The integrity test will stage a door fan testing rig inside one of the relay room doors. The testing rig will not block the relay room door from shutting. However, the relay room door does need to be manually held open for the duration of the test.

The door that will be temporarily manually held open is Door 56, U2 TURB BLDG TO RELAY RM.

Summary of 50.59 Evaluation

The proposed relay room leakage test includes measures to protect the design function of the relay room steam exclusion boundary and the design functions of the equipment located in the relay room. Therefore, this activity does not impact equipment operations, performance and reliability. Thus, there is no change to the frequency of an accident, likelihood of a malfunction, possibility of a new accident, or possibility of a malfunction with a different result.

The protective measures will ensure all relay room SSC remain in an operable condition such that any equipment operability assumptions used in dose analyses remain unchanged so there can be no increase in any accident consequences.

Therefore, with the protective measures employed, the test is consistent with Prairie Island's Current Licensing Basis and may be implemented without prior Nuclear Regulatory Commission approval.

50.59 Evaluation No. 1098 R0 - Unit 1 Cycle 28 Core Reload Modification

Description of Change

This activity will replace depleted fuel from the Unit 1 Cycle 27 reactor core with 56 feed (fresh) fuel assemblies and the rearrangement of used fuel assemblies of the 422 Vantage Plus (422V+) design. This will allow the Unit 1 reactor to produce power at its rated capacity in Unit 1 Cycle 28 for approximately 22 months. This activity is required because the fuel in the current core will be depleted to a state that no longer allows for full power operation. This evaluation is valid for operation of Unit 1 Cycle 28 in Modes 1 through 6.

Summary of 50.59 Evaluation

The UFSAR Chapter 14 evaluations performed by Westinghouse demonstrate that the Prairie Island Unit 1 Cycle 28 reload design and associated COLR do not result in the licensed safety limits for any accident being exceeded. The Cycle 28 design is consistent with the description of the core in the USAR. The core contains 121 fuel assemblies using a 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the UFSAR. The only change from Cycle 27 is the addition of new 422V+ fuel assemblies and the rearrangement of used fuel assemblies of the 422V+ design. This change results in an isotopic distribution of the core that changes the core physics parameters. The effect of these changes in the cycle physics parameters on cycle operation and accident analyses have been evaluated using NRC-approved methods discussed in T.S. 5.6.5.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore, the reload modification for Unit 1 Cycle 28 is consistent with Prairie Island's Current Licensing Basis and does not need prior NRC approval prior to implementation.

50.59 Evaluation No. 1101 R0 - Calculation CN-MRCDA-10-55 Evaluation of Updated Loads Due to Revised RV Support Stiffness

Description of Change

The proposed activity involves changing the design code of construction to a more recent edition of the code for the reactor vessel inlet nozzle safe-end and core support pads. The proposed activity is necessary because the newer edition of the code provides more specific criteria for the analysis being performed.

Summary of 50.59 Evaluation

Use of the more recent edition of the codes does not require prior NRC approval because the NRC has already codified this edition in 10 CFR50.55a(b)(1).

50.59 Evaluation No. 1102 R0 - Waste Gas Tank Rupture Dose Analysis

Description of Change

The proposed activity is to issue calculation 12400604-UR(B)-001, Revision 0A, Waste Gas Tank Rupture Dose Consequences, under EC 22469.

The Waste Gas Decay Tank rupture analysis contained within calculation 12400604-UR(B)-001, as described in USAR Section 14.5.3.1, is being updated to address the Prairie Island license extension from 40 to 60 years. The calculation determines the whole body dose consequences at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) based on the maximum amount of activity in a gas decay tank that could accumulate over the plant lifetime from operation with one percent of the rated core thermal power being generated by rods with clad defects. For all isotopes except Kr-85, this postulated amount of activity is taken to be one Reactor Coolant System equilibrium cycle inventory. The Kr-85 inventory is taken as the maximum activity that could have developed by the end of plant life. To address the Prairie Island license extension, the Kr-85 inventory must be recalculated based on a plant life of 60 years, which results in higher activity levels and dose rates at the EAB and LPZ. This calculation determines that the dose consequences resulting from a Waste Gas Decay Tank rupture are 4.32 REM at the EAB and 1.18 REM at the LPZ.

USAR Section 14.5.3 currently considers a waste gas decay tank (WGDT) rupture accident with the release of the radionuclide inventory at the end of 40 years of plant operation and will be updated to reflect 60 years of plant operation.

Summary of 50.59 Evaluation

Calculation 12400604-UR(B)-001 demonstrates that the resulting offsite dose consequences for a waste gas tank rupture after 60 years of operation were greater than the dose values that are currently in the USAR, thus making the completion of this calculation an adverse activity. However, the increase in the offsite dose values would not be more than minimal as defined by the 10% Rule discussed in NEI 96-07, Rev. 1. Therefore this activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR (basis question 3).

The consequences of a malfunction of an SSC important to safety are not changed because the calculation does not involve SSCs that are initiators of new malfunctions and no new failure modes are introduced (basis question 4).

The calculation does not involve any accident initiators or any physical changes to any SSCs important to safety or how they operate. Therefore, this activity has no impact on the frequency or likelihood of occurrence of an accident or a malfunction, nor does it create the possibility of an accident or a malfunction of a different type (basis questions 1, 2, 5, 6).

The calculation does not involve any design basis limits for fission product barriers (basis question 7).

The calculation methodology was the same as described in the USAR and therefore this activity does not involve a departure from an approved method (basis question 8).

50.59 Evaluation No. 1103 R0 - Control Room Envelope Special Test

Description of Change

The proposed activity is to perform an in-leakage test of the Control Room Envelope (CRE) with the control room (CR) chiller rooms' floor drain loop seals dry. The in-leakage test of the CRE is normally performed under the assumption that the loop seals are filled with water, thus forming a perfect seal between the Auxiliary Building Special Vent Zone (ABSVZ) and the CRE. The special test includes removing the water seals prior to performing in-leakage test. The loop seals will then be re-filled at the conclusion of the test.

The impacted SSC is the CRE which is portion of the Control Room Special Vent Zone (CRSVS).

The proposed activity will provide data to support a past operability evaluation. As discussed in CAP 01392548, Unacceptable preconditioning used to meet TS.5.5.16, SP 1449 (Rev 0), TRACER GAS TEST OF CONTROL ROOM, incorporated a step to add approximately 2 gallons of water to the floor drains prior to performing the in-leakage test. The intent of this step was to assure the loop seals in the floor drains were full of

water to prevent in-leakage through the floor drains. Addition of water to the floor drains prior to performing the Technical Specifications required tests constitutes preconditioning and invalidated the test results.

When the CRE was declared inoperable, staff implemented a mitigating action to meet required action B.1 of Condition B in TS 3.7.10. That mitigating action was to close the opening in the CRE by filling the floor drains. The mitigating action was then verified as meeting action B.2, ensuring the CRE occupant radiological exposures would not exceed limits, and the CRE occupants would be protected from chemical and smoke hazards. The plant is currently operating under the 90-day restoration limit imposed by action B.3.

ST 1449 temporarily removes the water from the floor drains, temporarily removing the mitigating action. The B.1 mitigating action will continue to be met by:

1) Installing a plumber's style plug in the floor drain; and

2) Administrative controls to open and close the loop seal CRE boundary.

This evaluation establishes that the administrative controls are adequate to meet the requirements of action B.2. Thus, the plant will remain in action B.3.

Summary of 50.59 Evaluation

The proposed activity does not affect the ABSVZ as the total openings in the ABSVZ will remain below 10 ft2. The proposed activity includes measures that ensure the CRE will be returned to its pre-test configuration should a DBA occur during its conduct. This ensures there would be no increase in dose to the CR occupants. Therefore, the test is safe and consistent with the current licensing basis.

The evaluation shows that the proposed activity has no effect on the probability or consequence of accidents or malfunctions. It does not create the possibility of an accident of a different type or a malfunction with a different result. There are no design basis limits for a fission product barrier exceeded, and there is no departure from a method of evaluation described in the Safety Analysis Report. Therefore, the test may be implemented without prior NRC approval.

50.59 Evaluation No. 1105 R0 - Unit 2 MSLB Containment Response With RSGs

Description of Change

The purpose of this evaluation is to implement a new calculation as the Analysis of Record for the Unit 2 containment response to a Main Steamline Break inside of containment. This activity is required due to replacement of the Unit 2 steam generators.

Summary of 50.59 Evaluation

This activity does not require prior NRC approval to implement as the new analysis used the Prairie Island current licensing basis NRC approved methodology and the results showed that the design limits as currently described in the Prairie Island Update Safety Analysis Report are met. In addition, this activity does not impact equipment operations, performance and reliability thus there is no change to the frequency of an accident, likelihood of a malfunction, possibility of a new accident, or possibility of a malfunction with a different result.

50.59 Evaluation No. 1106 R0 - Main Steam and Feedwater Break Thrust Reaction Load Methodology

Description of Change

The proposed activity involves changing a methodology for structurally qualifying the Unit 2 Reactor Coolant loop piping with the Replacement Steam Generators.

Summary of 50.59 Evaluation

Use of the software program to determine main steam and feedwater pipe break thrust reaction loads does not require prior NRC approval because the software calculates conservatively larger reaction load forces when compared to the hand calculation method used previously.

ENCLOSURE 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
CHANGES TO REGULATORY COMMITMENTS

Regulatory Commitment Change 12-01 – Change commitment wording for upgrading or installing penetration seal assemblies of the barrier separating the turbine building and auxiliary building

Commitment Source Document:
NSP Letter to NRC, 12/8/76, Comparison of Existing Fire Protection Provisions to SRP 9.5.1
NRC Letter to NSP, 11/21/78, Staff Positions and Staff Concerns
NSP Letter to NRC, 3/9/79, NSP Responses to Staff Positions
NRC Letter to NSP, 9/6/79, Fire Protection Safety Evaluation Report
NRC Letter to NSP, 4121/80, Fire Protection Safety Evaluation Report Supplement
NSP Letter to NRC, 10/24/80, Information Related to Fire Protection Modifications
NRC Letter to NSP 12/29/80, Fire Protection Safety Evaluation Report Supplement

Purpose for Original Commitment:
Upgrade existing or install new penetration seal assemblies in penetrations of the barrier separating the turbine building and auxiliary building. Of particular concern here is the portion of the barrier separating the 735 ft elevation of the Turbine Building (Fire Area 8) and the Auxiliary Building (Fire Area 60 and Fire Area 75)

Original Commitment Wording:
There are variations of the wording from document to document. The following is from the September 6, 1979, NRC FPSER, with only item (3) applicable to this commitment change evaluation. This is also reflected in item 027 of Commitment 72151:
"Penetrations in the barriers between the turbine building and auxiliary building will be sealed, and existing seals upgraded to provide 3-hour fire-rated seals."

Revised commitment wording:
Add the following at the end *of* Item 027, "Per Section 2.C(4) of the Facility Operating License for each unit, NSPM may make changes to the approved Fire Protection Program without prior approval of the Commission only If those changes would not adversely affect the ability to achieve and maintain safe shutdown In the event of a fire. FPEE-12-002 documents that unsealed main steam and feedwater penetrations PENF 1526, 1528, 1530, 1533, 1686, 1687, 1689, and 1692 on the 735 ft elevation would not adversely affect the ability to achieve and maintain safe shutdown In the event of a fire.

Justification for Change:
FPEE-12-002 documents that the lack of 3-hour lire-rated seals where main steam and feedwater penetrations PENF 1526, 1528, 1530, 1533, 1686, 1687, 1689, and 1692 penetrate the turbine and auxiliary building wall on the 735 ft elevation would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Postulated fires on either side of the barrier will not damage both the turbine slop valves in the turbine building and the main steam isolation valves in the auxiliary building.

Regulatory Commitment Change 12-02 – Change commitment wording for implementing NRC Generic Letter 96-05 and Periodic Verification Testing

Commitment Source Document:
NSP Correspondence dated March 17, 1997, from Joel Sorensen to US Nuclear Regulatory Commission, "Response to Generic Letter 96-05, Periodic Verification of Design Basis Capability of Safety Related Motor-Operated Valves"

Purpose for Original Commitment:
The purpose of the original commitment was to implement the NRC Generic Letter 96-05 AND the requirements of the Joint Owners' Group (JOG) Periodic Verification Testing Program (as described in MPR-1807, Revision 0)

Original commitment wording:
"The JOG periodic verification program will be used for Prairie Island safety-related MOVs to provide periodic verification that the valves are capable of performing their safety functions."

Revised Commitment Wording:
The JOG Periodic Verification (PV) program will be used in accordance with JOG Document MPR-2524-A for those valves that have been categorized as JOG Class A, B, or C for those valves currently in the Prairie Island GL 89-10/96-05 MOV program.

Justification for Change:
Per MPR-2524-A, Joint Owners' Group (JOG) Motor Operated Valve Periodic Verification Program Summary, all valves included within the scope of the GL 96-05 test program require classification into one of four categories: A, B, C, or D. If a valve is categorized as Class D, it has material combinations and/or system operating conditions that fall outside the scope of the original JOG testing program. As such, the JOG document (MPR-2524-A) cannot be used as justification for the establishment of the design basis for any Class D valves. The MSIV Bypass Motor Valves (MV-32045, MV-32047, MV-32048 and MV-32050) were the only valves in Prairie Island's GL 96-05 testing program that were categorized as Class D.

Regulatory Commitment Change 12-03 – Change commitment wording to remove Class D valves from JOG MOV PV Program

Commitment Source Document:
NSP Correspondence dated May 3, 1999, from Joel Sorensen to US Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Generic Letter 96-05 Program at Prairie Island Nuclear Generating Plant"

Purpose for Original Commitment:
The purpose of the May 3, 1999 commitment was to update the commitment to the JOG Program to be in accordance with MPR-1807, Revision 2.

Original Commitment Wording:
"Prairie Island will continue to participate in the JOG MOV Program consistent with Revision 2 of the JOG topical report and the NRC safety evaluation described above."

Revised Commitment Wording:
The JOG PV Program classifies each MOV into one of four classes: Class A, B, C, or D. Class A, B, and C valves are included within the scope of the JOG MOV PV Program and Class D valves are not. As such, the JOG PV Program will not be implemented for the Class D valve population which consists of the Main Steam Isolation Valve (MSIV) Bypass MVs (MV-32045, MV-32047, MV-32048 and MV-32050).

Justification for Change:
RIS-2011-13, Follow up to Generic Letter 96-05 for Evaluation of Class D Valves Under Joint Owners Group Motor-Operated Valve Periodic Verification Program, required that, if a station has any Class D valves and that station had committed to implement the JOG, a commitment change is required since, by definition, Class D valves fall outside the scope of the JOG testing program (MPR-2524-A).