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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
Renewed 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 one enclosure containing descriptions and summaries of safety evaluations for changes, tests, and experiments made under the provisions of 10 CFR 50.59 during the period of December 2015 to present. Since the last update, there have been no changes to regulatory commitments made within our Regulatory Commitment Change Process.

Summary of Commitments

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

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosure

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

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

REPORT OF CHANGES, TESTS, AND EXPERIMENTS - DECEMBER 2017

8 pages follow

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

REPORT OF CHANGES, TESTS, AND EXPERIMENTS - DECEMBER 2017

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. 1122, Rev. 1 – Heater Drain Tank Pump Drive Replacement (10/5/2016)

Description of Change

This activity covers replacement of the existing Heater Drain Pump (HDP) motors associated Regutron II speed control system equipment with new Variable Frequency Drives (VFDs). In addition the existing single Heater Drain Tank (HDT) level transmitter will be replaced with two new digital Guided Wave Radar (GWR) level transmitters. The existing Regutron II speed control system has become obsolete, component failures are higher, maintenance and repairs are extensive and troubleshooting is complicated. The introduction of software for the HDP VFDs and HDT level transmitters has introduced possible different behavior or failure modes that could affect the design functions. The replacement of the speed control system and the replacement of the level transmitters are separate elements of the change that do not meet the linking criteria and are therefore evaluated as separate activities.

Summary of 50.59 Evaluations

Based upon that the GWR technology has a good history of reliability, has no new failure modes with a different result, meets all applicable design and functional requirements, is more accurate in the operating conditions and process environment to which it is applied and the added redundancy of the two transmitters the installation of the two digital GWR HDT level transmitters is not considered to more than minimally impact the likelihood of an accident or malfunction and therefore does not require prior NRC approval for implementation. Based upon that the VFD technology has a good history of reliability with provisions for testing with acceptance criteria established to assure that the programmed parameters meet the applicable performance requirements, has no new failure modes with a different result, the new HDP motors are designed to be compatible with the VFD operation, and the VFD is designed for operation within the Turbine Building environment the installation of digital VFDs for HDP speed control is not considered to more than minimally impact the likelihood of an accident or malfunction and therefore does not require prior NRC approval for implementation.

ENCLOSURE 1

50.59 Evaluation No. 1125, Rev. 1 - USAR change for USAR Appendix L, Section L2.21 (9/9/2016)

Description of Change

The activity being evaluated is USAR change, which revises the frequency requirement to perform the manhole and pull box inspections in Section L.2.21. License Renewal Commitment #17 found in USAR Section - Appendix L states that PINGP will implement an Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program with features described in LRA Section B2.1.21. Inspections are changed from a 1-year frequency to a 5-year frequency. This change is being pursued because the current 1-year frequency for these inspections was identified as a potentially unnecessary burden on the site resources since no indications or evidence of water intrusion has ever been identified in either the 34+ years prior to starting the inspections or since beginning these inspections. This assessment is determining whether it is necessary or not to perform the inspections at a 1 year frequency.

Summary of 50.59 Evaluations

This evaluation concludes that prior NRC approval is not required based on the fact there is not more than minimal change in the frequency of occurrence of accidents and in the likelihood of occurrence of malfunctions due to the lengthening of inspection intervals. In addition, there is no increase in the consequence of an accident or malfunction, there is no possibility of an accident or malfunction of a different type since the probable accidents and malfunctions have already been evaluated in the USAR Section 14. This activity does not exceed or affect a DBLFPB. Plant specific operating experience shows no indications or evidence of water intrusion being identified in manholes or pullboxes in the 34+ years prior to or since starting these inspections.

50.59 Evaluation No. 1128, Rev 0 – Nuclear Fuel Burnable Absorber Transition to IFBA/Gadolina (5/26/2016)

Description of Change

The purpose of this evaluation is to incorporate the use of a new burnable absorber into the current licensing basis fuel design utilizing gadolinia. The new burnable absorber is in the form of a thin Boron-10 coating on the fuel pellets known as "Integral Fuel Burnable Absorber" or IFBA. This change is being made to improve fuel cycle economics.

Summary of 50.59 Evaluations

This activity to utilize Integral Fuel Burnable Absorber (IFBA) bearing fuel rods in the existing Prairie Island fuel assembly design does not require prior NRC approval. The analysis performed to support use of IFBA assures current licensing basis accident analyses acceptance criteria will continue to be met without changing any design basis limits for fission product barriers (DBLFPBs) or other relevant licensing basis safety analyses acceptance criteria for which the site is designed to withstand. Other than the fuel design changes, this change does not impact equipment operations, performance or 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 methods used to evaluate the new fuel configuration were determined not to be a departure from a Method of Evaluation as described within the Updated Final Safety Analysis Report.

50.59 Evaluation No. 1129, Rev. 0 - Change USAR required inspection method for special lifting devices (7/14/2016)

Description of Change

Per ANSI N14.6, special lifting devices are lifting devices for containers of radioactive materials which transmit the load from lifting attachments, which are structural parts of a container, to the hook(s) of an overhead hoisting system, excluding simple slings or chains. The current method of inspection endorsed by the USNRC for major load carrying items, including welds and other critical areas, is to perform dimensional testing, visual inspection, and non-destructive examination (NDE) evaluations using either magnetic particle testing (MT), or dye penetrant testing (PT) exams, which are surface evaluations.

The activity being evaluated is USAR change 01521758, which is to revise USAR Section 12 to permit an alternate method of inspecting special lifting devices. The proposed alternate method of inspection is to use AE NDE and, if required, supplemental examinations, including ultrasonic exams (UT), MT, or PT, in lieu of only MT or PT to examine major load-carrying welds and other critical areas of special lifting devices. AE NDE is described in detail in EPRI Report TR-107147. Essentially, AE NDE uses sensors placed strategically throughout the special lift device, to detect sound waves in the device as it loaded. The AE monitoring equipment can then filter the sound input received and isolate input that is generated due to propagating stress state pulses at a crack or overloaded location. The placement of multiple sensors allows for the source (e.g., a crack) to be located within the lifting device. As a result, AE NDE can easily detect and locate single or multiple cracks, but AE NDE cannot easily size and evaluate the indications discovered. For this reason, any indications that might be discovered through AE NDE would need to be uniquely identified, characterized, and evaluated further using a supplemental examination (UT, MT, or PT). Note that both AE and UT methods are volumetric examinations, whereas MT and PT are surface examinations.

Summary of 50.59 Evaluations

This evaluation concludes that prior NRC approval is not required based on the fact there is not more than a minimal change in the frequency of occurrence of accidents and in the likelihood of occurrence of malfunctions due to the change in inspection method to AE (supplemented, as appropriate, by UT, MT, or PT). In addition, there is no increase in the consequence of an accident or malfunction, there is no possibility of an accident or malfunction of a different type since the probable accidents and malfunctions have already been evaluated in the USAR Section 14. This activity does not exceed or affect a DBLFPB.

50.59 Evaluation No. 1131, Rev. 0 - Revise CRGT Guide Card MRP-227 Initial Inspection Schedule (9/23/2016)

Description of Change

The reactor vessel internals inspection program is changed to delay the initial control rod guide tube assembly guide card inspection. The current initial baseline inspection schedule, which was approved by the NRC in license renewal submittals, calls for an initial VT-3 examination no later than two refueling outages from the beginning of the license renewal period. The revised initial baseline inspection schedule calls for an initial VT-3 examination based upon more recent industry guidance in WCAP-17451-P, "Reactor Internals Guide Tube Wear- Westinghouse Domestic Fleet Operational Projections", and crediting Prairie Island Units 1 and 2 reactor upper internals replacement in the mid-1980s.

Summary of 50.59 Evaluations

The proposed change has a no more than minimal effect on accidents and malfunctions previously evaluated in the UFSAR. The original schedule was approved based upon its consistency with NRC approved MRP-227-A Rev. 0. MRP-227-A recommended inspections no later than two refueling outages from the beginning of the license renewal period. The recommendation did not consider the fact that some plants had replaced their upper internals. Prairie Island upper internals were replaced in 1986. An upper internals 40-year life would be reached in 2026.

WCAP-17451-P Section 5 calls for an initial inspection after 35.3 EFPY on the new upper internals. A 40-year life, plus or minus two refueling outages, is roughly equivalent to 35.3 EFPY. The aging mechanisms for the guide cards are gradual. Therefore, the delay in the initial baseline inspection to conform to the recommendations ofWCAP-17451-P Section 5 does not create more than a minimal increase in the likelihood of a malfunction of the control rods.

50.59 Evaluation No. 1132, Rev. 0 - Unit 1 Cycle 30 Core Reload Modification (10/20/2016)

Description of Change

Replace depleted fuel from the Unit 1 Cycle 29 reactor core with 52 feed (fresh) fuel assemblies, and one reinsert fuel assembly from the spent fuel pool last used in Unit 2 Cycle 28 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 30 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 30 in Modes 1through 6.

Summary of 50.59 Evaluations

The UFSAR Chapter 14 safety analysis performed by Westinghouse demonstrate that the Prairie Island Unit 1 Cycle 30 reload design and associated Core Operating Limits Report do not result in the licensed safety limits for any accident being exceeded. The Cycle 30 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 physical change from Cycle 29 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 and the cycle does not exceed any fuel burn up limits. The change described above was evaluated against the eight criteria of 10CFR 50.59(c) (2) and none of the criteria were met. Therefore, the reload modification for Unit 1 Cycle 30 is consistent with Prairie Island's Current Licensing Basis and does not need NRC approval prior to implementation.

50.59 Evaluation No. 1133 Rev. 0 - Manual Actions Required to Align Unit 1 Component Cooling System to the 122 Spent Evaluation Title: Fuel Pool Heat Exchanger (9/27/2016)

Description of Change

Operator actions will be added to 1C14, Component Cooling System- Unit 1 Rev 43, to momentarily maintain the control switches in the OPEN position for the fast-acting valves CV-39153 & CV-39154 when aligning the Component Cooling (CC) system to provide cooling flow through the 122 Spent Fuel Pool (SFP) Heat Exchanger (HX). Operator actions will also be added to 1C14 to momentarily maintain the control switches in the OPEN position when starting or stopping the standby Unit 1 CC Pump (11 or 12 CC Pumps) for up to one (1) minute.

Summary of 50.59 Evaluations

The evaluation found that the proposed procedure changes was determined to not result in more than a minimal increase in the frequency of occurrence or the consequences of an accident, in the likelihood or consequences of occurrences of a malfunction of a system important to safety, or in the possibility of an accident of a different type or a malfunction with a different result. It was also found to not result in a design basis limit for a fission produce barrier. This activity does not involve any methods of evaluation described in the Updated Safety Analysis Report (USAR).

50.59 Evaluation No. 1134, Rev. 0 - EVALUATION OF LOOSE PART IN PI UNIT 1 REACTOR VESSEL PAST 1R30 (11/10/2016)

Description of Change

During Refueling Outage 1 R29, a socket head cap screw was unaccounted for and assumed left behind in the Unit 1 Reactor Vessel. Two washers that had been connected to a submarine with this cap screw were found during Foreign Object Search and Retrieval activities prior to core reload. The cap screw was not known to be missing at the time and is thus presumed to be left behind in the reactor vessel below the fuel assemblies. The proposed activity being evaluated is the use-as-is non-conforming condition of the presence of the socket head cap screw in the Reactor Coolant System during operation. The location, size, and mass of the socket head cap screw will not impact safe operation.

Summary of 50.59 Evaluations

Operation with the cap screw will cause less than a minimal increase in the frequency or likelihood of occurrence of RCS pressure boundary ruptures or loss of integrity. Even if the cap screw were to position itself directly in line with the flow path through the fuel assemblies, the size of the flow restriction is so small that it will have a less than minimal effect on RCS flow through fuel assemblies. There is no increase to the consequences of an accident or malfunction. All design limits continue to be met even with the cap screw in this part of the RCS. No accidents of a different type are created by RCS operation with this cap screw. There is no possibility of a malfunction of any components in the RCS with a different result due to the cap screw. Also, this activity does not affect any methods of evaluation as described in the USAR

50.59 Evaluation No. 1136, Rev 0 EC 25124 – NFPA 805 – PINGP STATION – 12&22 DDCLP, D1 BKR 15-2, 1R XFMR (LAR Items 14, 34, 35), (3/23/2017)

Description of Change

This evaluation covers a portion of the changes to be installed under EC 25124 to support transition to the NFPA 805 licensing basis. The DDCLP control circuits are being modified as directed by L-PI-16-090 Table S-2 Item 14 to ensure that both pumps will start automatically on low CL system header pressure after a fire in the control room (FA-13) and/or relay and cable spreading room (FA-18) has required evacuation of the control room. The changes will isolate the control circuit from these fire areas and all portions required for operation of the pumps will be in the screen house or the AFW pump rooms. Fusing and protective relaying will be installed that will completely isolate the portion of the circuit that is in FA-13/18. A new circuit pathway is being created that will automatically start the DDCLPs on low CL header pressure. Also, relays are being relocated to ensure that the scavenging and combustion air dampers and the screen house exhaust fans will operate automatically. Thus the currently credited manual actions in the screen house to locally start the DDCLPs, open the dampers, and start an exhaust fan, and the actions in the D1 room to check CL system pressure are no longer required and are being removed from calculations GEN-PI-026 (minor revision 6J) and GEN-PI-055 (minor revision 1L) and from procedure F5 Appendix B and F5 Appendix E.

Summary of 50.59 Evaluations

The DDCLPs provide supporting functions to remove heat from plant equipment and to provide an auxiliary supply of water, and thus are not initiators to any accident or event and thus the changes made to the DDCLP control circuits by EC 25124 for LAR 14 do not increase the likelihood of occurrence of an accident or event evaluated in the USAR, nor do they create the possibility of an accident or event of a different type than any previously evaluated in the USAR. The components added to the control circuits are typical and consistent with the existing circuit components and do not reduce the reliability of the DDCLPs. As documented in EC 25124, the control circuit changes are an improvement in terms of the ability to withstand the potential circuit failures caused by a FA-13/18 fire, and the new circuit design allows for fewer and simpler manual actions in combination with automatic actuation circuitry that will not be affected by the fire. Therefore the EC 25124 changes don't increase the likelihood of a malfunction of a DDCLP to perform its design basis function or its function to support plant fires. Furthermore, this justification also allows the conclusion to be drawn that these changes will not increase, or even affect the consequences of an accident or a malfunction of equipment important to safety that has been previously evaluated. The circuit changes do not create any new failure modes and thus the possibility for a malfunction of a DDCLP with a different result is not created. The EC 25124 changes do not involve any fission product barrier, and all calculations that involve methods of evaluation were performed consistent with the USAR requirements (per SCR 5229). In conclusion, the changes that EC 25124 is making to the DDCLP control circuitry and the elimination of the credited manual actions as identified in F5 Appendix B and F5 Appendix E for the response to a fire that requires an evacuation of the control room can be implemented without NRC approval. Note that this conclusion is based on an evaluation against the Appendix R licensing basis but also requires the transition to NFPA 805 to have been completed prior to installation.

50.59 Evaluation No. 1137, Rev. 0 - New Manual Actions to Support Battery Room Temperatures (7/17/2017)

Description of Change

A new room heat-up analysis for a Design Basis Accident (DBA) demonstrates a potential need for new operator actions to support the design function of the Safeguards Batteries. The new operator actions are as follows: Operator actions previously added to C18.1, "Engineered Safeguards Equipment Support Systems", to open all Safeguards Battery Room Doors (DOOR 224, DOOR 225, DOOR 226, DOOR 227, and DOOR 228) prior to the temperatures in the room reaching 120°F will be repurposed as permanent (versus a compensatory measure) for a DBA. Operator action will also be added to reset the Instrument Inverter AC input breaker (CB-401) for the Unit 1 Inverters if the input breaker tripped during the Emergency Diesel Generator start-up and load sequencing (CAP 01270104). The change will be implemented per PCR 6PCR01560860. The PCR makes three main changes to the procedure. 1. The PCR modifies an existing note before Step 5.24.5 to clarify the approximate time operations has between when they would get notification of 118°F temperature in the

Battery Rooms and when the 120°F limit would be reached for a Design Basis Accident with a Loss of Offsite Power 2. The PCR modified Step 5.24.5 to remove the reference to the OPR and the word compensatory. This will help to clarify that the sub-steps in Step 5.24.5 are being repurposed as permanent actions and will no longer be compensatory actions for purposes of a Design Basis Accident. 3. The PCR also adds new Step 5.24.6 to check to see if DC input to any Unit 1 Inverter reads greater than zero (0) amps. If so, then the step provides direction to reset the applicable AC Input Breaker to inverters 11, 12, 13, 14, 17, and 18.

Summary of 50.59 Evaluations

The new operator actions support the ability of the Safeguards Batteries (11 BATT, 12 BATT, and 22 BATT to perform their design function to carry expected shutdown loads following a plant trip, and a loss of AC battery charging power for a period of 1 hour without battery terminal voltage falling below the required minimum in the event of a Design Basis Accident.