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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, (NSPM) doing business as Xcel Energy, 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.

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Enclosures (2)

cc: Administrator, Region III, USNRC
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ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS, AND EXPERIMENTS – DECEMBER 2011

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, (NSPM) without prior Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR 50.59.

50.59 Evaluation no. 1025 Revision (R) 5 - 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 15963 which evaluates the effects of a zebra mussel treatment, per D104.1, on plant equipment with consideration to the current mussel population. Additionally, discussion of plant traveling screens has been revised.

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 (Structures, Systems, and Components) important to safety previously evaluated in the USAR (Updated Safety Analysis Report) or any pending submittal. It has also been determined that there will be no affect 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. 1057 R0 – U1 & U2 Replacement of Rad Monitors R-11/R-12

Description of Change

This evaluation addresses Engineering Change (EC) 7776 to replace the existing R-11/R-12 Containment/Shield Building Air Particulate and Gaseous radiation effluent monitoring channels for Units 1 and 2. This activity installs improved radiation monitoring skid systems.

The primary scope of this activity is to replace the existing radiation monitor skids, Control Room switches, indication, and analog readout meters with reliable, safe and proven equipment including digital display units and associated software. This activity also includes design changes to assure more accurate indications.

Summary of 50.59 Evaluation

The replacement of the existing R-11/R-12 radiation monitoring equipment with new improved equipment does not introduce the possibility of a change in the frequency of an accident because this equipment is not an initiator of any accident, and no new (accident) failure modes are introduced. These Radiation Monitors are not relied upon to mitigate the consequences of any design basis accident.

This activity does not introduce the possibility of a change in the likelihood or consequences of a malfunction of equipment because it was performed in accordance with applicable Regulatory and Industry codes & standards. This activity does not create a malfunction with a different result because the fit, form and function has not changed, and the simplicity and reliability of the design and software.

This change does not change a method of analysis described in the USAR and does not involve a design basis limit for a fission product barrier.

Therefore the proposed activity does not require prior NRC approval.

50.59 Evaluation no. 1072 R3 – U1 Cycle 26 Reload

Description of Change

This activity will replace depleted fuel from the Unit 1 Cycle 25 reactor core with 48 fresh fuel assemblies. This will allow the Unit 1 reactor to produce power at its rated capacity in Unit 1 Cycle 26 for approximately 18 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.

Revision 3 to this 50.59 evaluation will incorporate a revision to the COLR for Unit 1 Cycle 26 to correct an inaccurate shutdown margin value for Mode 2 listed in Table 1.

Summary of 50.59 Evaluation

The USAR Chapter 14 evaluations performed by Westinghouse demonstrate that the Prairie Island Unit 1 Cycle 26 reload design and associated COLR do not result in the licensed safety limits for any accident being exceeded. The Cycle 26 design is consistent with the description of the core in the USAR, including pending changes pursuant License Amendment *192/181* which approved use of the 422 Vantage Plus (422V+) fuel assembly. 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 25 is the distribution of new 422V+ fuel assemblies with used assemblies of the Optimized Fuel Assembly (OFA) design. This change results in a redistribution of the isotopic distribution of the core that changes the core physics parameters of the reactor. 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 1 Cycle 26 is consistent with Prairie Island's Current Licensing Basis

50.59 Evaluation no. 1073 R0 – Change in Methodology to GOTHIC Version 7.2a for Compartment Environmental Response Outside Containment

Description of Change

The change being evaluated is the use of GOTHIC 7.2a for compartment environmental responses outside of Containment. The SSCs impacted by the change are the compartments outside of Containment and the enclosed systems and components. The applicable SSC (compartments) include but are not limited to those located in the Auxiliary and Turbine Buildings. The current licensing basis at Prairie Island permits the use of GOTHIC 7.2a inside Containment per Amendment No. 192 (181) to Facility Operating License No. DPR-42 (DPR-60), Prairie Island Nuclear Generating Plant, Unit 1 (2). The licensing basis also permits the use of GOTHIC 6.1 in Auxiliary Building compartments, as evaluated in SE 1039. There is no current licensing basis documented regarding a methodology for analysis of plant compartments outside of the Containment and Auxiliary Building. The purpose of this activity is to evaluate whether NRC pre-approval is required for a change in methodology for compartment environmental responses for plant compartments outside Containment using the most current version of GOTHIC, version 7.2a. The "change" is a potential change in approved methodology only. [Note: The results of calculation ENG-ME-767 have been screened by Screening #3398.] In the context of this evaluation, user options used to select the specific theoretical formulas used within the code are considered to be part of the methodology. The design, design inputs, and design basis requirements are not affected.

Summary of 50.59 Evaluation

This evaluation concludes that prior NRC approval is not required to use GOTHIC 7.2a to analyze environmental responses for compartments outside of Containment. GOTHIC 7 (version 7.1 patch 1) was approved for use at Prairie Island per license Amendments 171 & 161 (WCAP-16219-P). License Amendment Nos. 192 & 181 extended the technical justification of WCAP-16219-P to GOTHIC 7.2a for inside Containment applications. A tabular summary of the differences in specified user options between the use of the GOTHIC 7.2a for inside and outside Containment applications was developed and evaluated. This evaluation concludes that there are no significant differences between the proposed activity and what was previously approved by the NRC.

50.59 Evaluation no. 1074 R0 - Waste Gas Tank Rupture Dose Analysis

Description of Change

Shaw Stone & Webster (SSW) calculation 12400604-UR(B)-001 is to be accepted as the analysis of record for the waste gas system radiological design basis accident (DBA) analysis via EC 15424. This analysis documents the offsite dose consequences for a rupture of the volume control tank and a waste gas decay tank. The current analysis of record for the waste gas system radiological DBA analysis exists only in the USAR. Validation of licensing basis activities conducted in support of the Measurement Uncertainty Recapture (MUR) Project was not able to validate the USAR information against a calculation of record. In addition, this validation effort was not able to validate that this accident analysis had been evaluated for impact due to changes to fuel type and fuel cycle length since original licensing. It is a vulnerability to not have a technical document which supports the design and licensing basis of plant. Therefore, SSW was contracted to complete a confirmatory analysis of the existing licensing basis utilizing the methodology and input described in the USAR and also to complete a design validation analysis based on the USAR methodology which incorporates the fuel cycle management allowances. The SSW analysis documents an increase in offsite dose consequence as compared to the recorded consequence in the USAR.

Summary of 50.59 Evaluation

Calculation 12400604-UR(B)-001 demonstrated that the resulting offsite dose consequences for a waste gas tank rupture were greater in some cases 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. 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. 1075 R0 - Steam Generator Tube Rupture Dose Analysis

Description of Change

Shaw Stone & Webster (SSW) calculation 12400604-UR(B)-003 is to be accepted as the analysis of record for the radiological consequences of a Steam Generator Tube Rupture design basis accident (DBA) via EC 15505. This analysis documents the offsite dose consequences for a steam generator tube rupture. The current analysis of record for the radiological consequences of this DBA exists only in the USAR. Validation of licensing basis activities was not able to validate the USAR information against a calculation of record. In addition, this validation effort was not able to validate that this accident had been evaluated for impact due to changes to fuel type and fuel cycle length since original licensing. It is a vulnerability to not have a technical document which supports the design and licensing basis of the plant. Therefore, SSW was contracted to complete a confirmatory analysis of the existing licensing basis utilizing the methodology and input described in the USAR and also to complete a design validation analysis based on the USAR methodology which incorporates the fuel cycle management allowances. The SSW analysis documents an increase in offsite dose consequence as compared to the recorded consequence in the USAR.

Summary of 50.59 Evaluation

Calculation 12400604-UR(B)-003 demonstrated that the resulting offsite dose consequences for a Steam Generator Tube Rupture were greater in some cases 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 increase in the consequences of an accident previously evaluated in the UFSAR (basis question 3).

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

The calculation does not impact any accident initiators or make any physical changes to 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. Fission product barriers are part of the analysis, but no limits are changed (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. 1076 R1 - Unit 2 Cycle 26 Core Reload

Description of Change

This activity will replace depleted fuel from the Unit 2 Cycle 25 reactor core with 56 fresh fuel assemblies. This will allow the Unit 2 reactor to produce power at its rated capacity in Unit 2 Cycle 26 for approximately 20 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 also changes the evaluation methodology used for Main Steam Line Break containment response. The new method is GOTHIC, an NRC approved method.

Revision 1 to this evaluation will install a new COLR (Core Operating Limits Reports) for Unit 2 Cycle 26 that includes W(z) penalty factors for use in part-power equilibrium flux maps. These part-power penalty factors were developed for specific power levels and will provide a more accurate prediction of possible non-equilibrium power shapes for the following 31 days.

Summary of 50.59 Evaluation

The UFSAR Chapter 14 evaluations performed by Westinghouse demonstrate that the Prairie Island Unit 2 Cycle 26 reload design and associated COLR do not result in the licensed safety limits for any accident being exceeded. The Cycle 26 design is consistent with the description of the core in the UFSAR, including pending changes pursuant to Unit 2 License Amendment 181 which approved use of the 422 Vantage Plus (422V+) fuel assembly. 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 25 is the distribution of new 422W fuel assemblies with used assemblies of the Optimized Fuel Assembly (OFA) design. This change results in a redistribution of the isotopic distribution of the core that changes the core physics parameters of the reactor. 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 26 is consistent with Prairie Island's Current Licensing Basis.

50.59 Evaluation no. 1077 R0 - Removal of NRC Commitments 0001028 and 0001029 for non-intrusive RCS (Reactor Coolant System) level indication during reduced inventory Operations

Description of Change

This activity removes the commitments to the NRC to have the RCS non-intrusive RCS level indication (which are the ultrasonic level instruments) during reduced inventory operations. Removal of these commitments will result revisions in procedures I C4.1, 2C4.1, I D2, 2D2, 1 D2.1 AND 2D2.1. This evaluation is specifically written to evaluate the adverse impact of removing a diverse method of RCS level indication.

Summary of 50.59 Evaluation

This change is considered adverse because the reduction of redundant diverse methods to detect reactor coolant level during reduced RCS inventory operation.

The proposed activity removes a diverse method to monitor RCS level during reduced RCS inventory evolutions by the removal of the commitment (and the changes to the associated implementing procedures) to have a non-intrusive method of monitoring RCS level during reduced inventory. The requirement for the monitoring of the RCS level during reduced inventory is to have 2 independent methods that provide continuous control indication. PINGP meets this with one of two trains of ERCS (Emergency Response Computer System) d/P Level instruments and one train of Refueling Canal level instrument to meet the requirements of Generic Letter 86-12 and 88-17.

The event which occurred at PINGP on February 20, 1992, involved the electronic reactor vessel water level instruments, which were intended to provide level indication in the control room on the Emergency Response Computer System, were in service but both showed failed.

The revisions to procedures 1C4.1, 2C4.1, 1D2, 2D2, 1D2.1 AND 2D2.1 remove the requirement for the ultra-sonic level instrumentation. The remaining instruments must be in service to assure the independent methods to provide monitoring of RCS level during reduced inventory.

A common mode failure for the two trains of ERCS d/P level instruments and one train of Refueling Canal level instrument is the only malfunction that could credibly occur as a result of removing the requirement to have the non-intrusive method to determine RCS level.

ERCS DP level indication A or B is the first source of RCS level indication (ERCS points 2L0460A and 2L0470A) as indicated on the ERCS Reduced inventory screen. The second independent source is level indicator LI-41972 located on the control board. The ERCS points originate from level transmitters 2LT-434 (21 RCS NARROW RANGE LVL XMTR) and 2LT-435 (22 RCS NARROW RANGE LVL XMTR). The transmitters have independent non-safety related power supplies. The control board indication is via level transmitter 24128 (2 RX RFLG CANAL LVL XMTR). Drawing XH-1001-3 shows the three level transmitters described above, all having separate pressure taps to the RCS. The ERCS points described above as well as the refueling canal level instrument are in the current revision to 2D2 as instruments to be used in accordance with Figure C1-40 for the reduced inventory operation. Independence of the ERCS DP and Refuel Canal level indication is maintained through venting of the RCS atmosphere and the use of gravity draining of the RCS. Forced draining of the RCS can provide an environment which can lead to over pressure of the RCS leading to erroneous indication of the ERCS DP and Refuel Canal level transmitters. Currently 1D2, 2D2, 1C4.1 and 2C4.1 contain steps that vent the RCS to atmosphere and accomplish draining of the RCS via gravity. Procedures 1D2.1 and 2D2.1 provide a vent path to the atmosphere through the pressurizer manway or both pressurizer safeties. Drain down is again accomplished via gravity.

These 2 independent sources, ERCS and the control board indication, satisfy the Generic Letter 88-17 requirements and do not have a common mode failure for a single failure of the compensation for RCS pressure.

The removal of NRC commitments 0001028 / 0001029 (and the changes to the associated implementing procedures) to have a non-intrusive method of monitoring RCS level during reduced inventory is considered to be the removal of excess indication (beyond requirements) and therefore considered minimal.

50.59 Evaluation no. 1078 R0 - Fuel Handling Accident (FHA) Dose Results Applying AST (Alternative Radiological Source Terms) Methodology (RG (Regulatory Guide) 1.183) In Support of Heavy Bundle Fuel (V422V+) Project

Description of Change

During the Heavy Bundle Fuel project, the analysis of record (AOR) for the Fuel Handling Accident dose consequence analysis (GEN-PI-051, Rev. 1) was being reviewed for revision to support the project. During this review, it was found that the AOR was based on a non-conservative maximum radial peaking factor. The analysis used a value of 1.65 as documented in USAR Section 14.5.1.1 (Page 14.5-4), USAR Table 14.5-1, and USAR Appendix D, Table D.3-1. Since the FHA dose analysis uses the AST methodology, it follows the guidance of Reg Guide 1.183 which states that the radial peaking factor used should come from the facility's core operating limits report (COLR) or technical specifications. The PINGP COLR for both units uses a maximum radial peaking factor of 1.77. Review of the LAR supported by the AOR demonstrated that no exceptions were taken to the analysis guidance provided in Appendix B of RG 1.183. Therefore, it is concluded that the value of 1.77 should have been used for the analysis.

The specific change being evaluated is a change to the underlying analytical bases demonstrating the ability of the site to meet dose consequence regulatory limits for a Fuel Handling Accident in accordance with 10 CFR 50.67 and Reg Guide 1.183. The above mentioned non-conservatism was resolved by the performance of a new calculation by Fauske & Associates, Inc (Calc *FAI/07-63)* using the maximum radial peaking factor value. In addition to the change in radial peaking factor, a revised core inventory reflects the use of heavy bundle fuel (V422V+) was also integrated into the calculation.

Summary of 50.59 Evaluation

Calculation FAI/07-63 demonstrated that an increase in the radial peaking factor resulted in offsite and control room dose consequences for a Fuel Handling Accident (FHA) that are greater than the dose values currently in the USAR. This dose increase makes the completion of this calculation an adverse activity. However, the increase in the offsite and control room dose values did not result in a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR (basis question 3).

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

The calculation does not impact any accident initiators or make any physical changes to 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. Fission product barriers are part of the analysis, but no limits are changed (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. 1079 R8 - Strategic Water Chemistry Plan for PINGP Primary System Chemistry

Description of Change

The activity being evaluated is the changing of the Reactor Coolant System's (RCS) target lithium concentration from current value of 2.2 ppm to 3.5 ppm. This change in target lithium concentration would be applicable after the first 150 megawatt day per metric ton of uranium (MWD/MTU) of operation for a given fuel cycle. This change could impact corrosion phenomena of RCS materials including general corrosion, primary water stress corrosion cracking, and fuel cladding oxidation. In addition, increasing the cycle average lithium concentration will slightly increase the tritium production in the RCS coolant. This change is being made to reduce the general corrosion rates of RCS materials by allowing the plant to operate with a higher RCS pH earlier in the fuel cycle than can be achieved with the current target lithium concentration. Reducing general corrosion should reduce 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 slight increase in tritium production is bounded by the current evaluation contained in the Updated Safety Analysis Report. 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.

Therefore the proposed activity does not require prior NRC approval.

50.59 Evaluation no. 1080 R0 - Change Licensing Basis for RCS Leakage Detection Capability from 1 GPM in 1 Hour to 1 GPM in 4 Hours

Description of Change

As the Reactor Coolant System radioactivity levels decreased due to improved fuel performance and reduced general corrosion, the ability to detect a leak based on radio nuclides released into containment decreased. Thus it is no longer practical for the leak detection system to be able to detect a one gpm leak in one hour called for in Prairie Island's Licensing Basis. This evaluation justifies changing the Prairie Island's leak detection system's Licensing Basis from being able to detect an one gpm leak in one hour to being able to detect a one gpm leak in four hours and clarifies that the time to detect a leak, i.e. the 4 hours, does not include the time for the leaking fluid to travel from the source to the detection site of the instrument.

Summary of 50.59 Evaluation

The applicable criteria for the leak detection system capability listed in Section 5.7 of NUREG-1 061, Volume 3 "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks" calls for a leak detection system with a sensitivity capable of detecting an unidentified leakage rate of one gpm in four hours. Since the proposed change continues to meet this criterion, there is less than a minimal increase in the frequency of occurrence of an accident. The change does not involve any changes to equipment and thus there is no change to the likelyhood or results of a malfunction. The leak detection system only performs a monitoring function and thus does not affect the consequences of, or introduces a different type of, accident or malfunction. The proposed activity does not involve a design basis limit for a fission product barrier or method of evaluation.

Therefore the proposed activity does not require prior NRC approval.

50.59 Evaluation no. 1083 R1 - Manual Actions Required to Restart Battery Chargers

Description of Change

The purpose of Revision 1 of this Evaluation is to address two additional considerations. The first consideration discusses the environmental considerations in which the operator must travel through to get to the battery rooms. This concern was identified per AR01266968. From analysis, EC EVAL (Evaluation) 17495 and CAP 01266968 provided that environmental conditions the operator would travel through when performing compensatory measures would be approximately 175°F with 100% humidity. Because of the environment, the battery charger watch instructions were enhanced.

The second consideration provides compensatory measure guidance for the battery watch operator when the portable battery charger is installed in place of a station battery charger. Compensatory measures discussed in this 50.59 EVAL were revised to include the Portable Battery Charger when installed in place of a station battery charger.

OPR 01238842-01 and OPR 01250561-02 are also revised concurrently with the evaluation to reflect the changes.

Activity Description:
OPR 01238842-01 Rev. 2 and OPR 01250561-02 Rev. 1 require compensatory measures in the form of new operator actions to support the design function of the Battery Chargers. The compensatory measures are as follows:

- Use of 1[2] C20.9 AOP3(4) Attachment B as a new operator action to support the design function of the Battery Chargers.
- Notation of the one hour time constraint in an existing AOP (1C20.9 AOP3(4) – Failure of 11(12) Battery Charger, 2C20.9 AOP3(4) - Failure of 21(22) Battery Charger) step that directs Operations personnel to restart the affected Battery Charger in the event of a Battery Charger lockup.
- The addition of a step to the end of the SI (Safety Injection) Alignment Verification section of the Reactor Trip or Safety Injection (Attachment L of procedure 1E-0 and 2E-0) to check proper operation of the Unit 1 and Unit 2 Battery Chargers (previously evaluated per 1083 Rev 0).

Summary of 50.59 Evaluation

The proposed compensatory measures ensure the battery chargers are restored during a DBA, if necessary, within one hour. The station batteries have "been sized 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." Restoration of the battery chargers within one hour allows the chargers to supply the 125 VDC System and maintain the battery in a charged condition.

The proposed compensatory measures have been evaluated per NRC INSPECTION MANUAL, PART 9900, TECHNICAL GUIDANCE, OPERABILITY DETERMINATIONS &

FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY, APPENDIX C, SECTION C.5 AND IN 97-78, CREDITING OF OPERATOR ACTIONS IN PLACE OF AUTOMATIC ACTIONS AND MODIFICATIONS OF OPERATOR ACTIONS, INCLUDING RESPONSE TIMES.

The proposed compensatory measures may be implemented without prior NRC approval.

50.59 Evaluation no. 1084 R0 - Response Time for R-11 Particulate Radiation Monitor to leaks of 1.0 gpm and 0.2 gpm

Description of Change

Calculation ENG-ME-792 Rev. 0 determines a revised response time for 1R-11 and 2R-11 to a step increase in leakage from the RCS, based on updated coolant activity levels, in order to show compliance with the design basis requirement to detect a 1.0 gpm leak within 1 hour. The calculation also shows the ability to detect a 0.2 gpm leak within 4 hours.

The previous analytical basis for the R-11 instrument response times to RCS leaks is contained in the document "Report to USNRC, Coolant Leakage Detection System Performance at the Prairie Island Nuclear Generating Plant", March 31, 1976. The actual calculation could not be located, however, the 1976 report from the NRC summarized the methodology and conclusions of the original calculation, This report is incorporated by reference in Section 6.5.1 of the USAR. ENG-ME-792 Rev 0 supersedes the detector response information contained in that report, which was based on actual measured coolant activity levels from December 10, 1975, and which are not conservative relative to current plant operating coolant levels.

R-11 was replaced on EC 7776 in 2010. In order to more accurately predict the instrument response time, more recent guidance on calculating response time was used - ISA 67.03-1982, "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection." This guidance has been endorsed in NUREGICR-6861 and has been incorporated into RG 1.45, Rev. 1. This guidance introduced three changes to the way the instrument response time was calculated when compared to the original calculation performed in 1976. These changes are discussed in more detail in later sections of this document. This evaluation looks at these differences to determine if NRC approval is required for the modified methodology.

1OCFR5O.59 Screening 3668 was performed on this activity. The only portions that did not screen out were those that had to do with the change of an element of the methodology, namely the plate out factor vs. flash fraction. This is discussed in more detail further in this evaluation. This 1OCFR5O.59 evaluation is only evaluating this one item since all others screened out in Screening 3668. Only criterion viii will be addressed in this evaluation.

Summary of 50.59 Evaluation

This evaluation has determined that no *NRC* approval is required to change the element of the methodology as described herein, namely the use of the plate out factor instead of the ft ash fraction as used in the original analysis described in the *USAR*.

50.59 Evaluation no. 1085 R0 - Manual Actions Required to Maintain Battery Room Temperatures

Description of Change

OPR 01265904-01 Rev. 0 requires a compensatory measure in the form of operator actions to support the design function of the Safeguards Batteries. The compensatory measure is as follows:

Operator action will be added to C18.1, Engineered Safeguards Equipment Support Systems, Rev. 29 to open all Safeguards Battery Room Doors (DOOR 224, DOOR 225, DOOR 226, DOOR 227, DOOR 228) prior to the temperatures in the room reaching 120°F.

The change will be implemented per TCR# 029-A to C 18.1 Rev. 29. TCR# 029-A adds Step 5.24.5 to contact security, establish a continuous fire watch, and block open Door 224, Door 225, Door 226, Door 227, and Door 228 when any Safeguards Battery Room temperature reaches 118°F.

A note preceding Step 5.24.5 was also added to provide clear direction to operations staff that Door 224 and Door 225 cannot be blocked open until [both Units MSIVs (Main Steam Isolation Valve) and MSIV bypass valves are closed OR when conditions no longer exist for a HELB to occur] and [the 695' Turbine Building temperatures are less than 118°F].

The doors that will be blocked open are:

- DOOR 224, U2 TURB BLDG TO 21 BATT RM
- DOOR 225, U1 TURB BLDG TO 11 BATT RM
- DOOR 226, 22 BATT RM TO 21 BATT RM
- DOOR 227, 11 BATT RM TO 12 BATT RM
- DOOR 228, 12 BATT RM TO 22 BATT RM

Summary of 50.59 Evaluation

The compensatory measure ensures the Safeguards Batteries (11 BATT, 12 BATT, 21 BATT, 22 BATT) can perform their design function in the event of a LOOP (Loss of Offsite Power) or SBO (Station Blackout). The batteries have been sized 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.

The compensatory measure is being evaluated per NRC INSPECTION MANUAL, PART 9900, TECHNICAL GUIDANCE, OPERABILITY DETERMINATIONS & FUNCTIONALITY ASSESSMENTS FOR RESOLUTION OF DEGRADED OR NONCONFORMING CONDITIONS ADVERSE TO QUALITY OR SAFETY, APPENDIX C, SECTION C.5 AND IN 97-78, CREDITING OF OPERATOR ACTIONS IN PLACE OF AUTOMATIC ACTIONS AND MODIFICATIONS OF OPERATOR ACTIONS, INCLUDING RESPONSE TIMES.

50.59 Evaluation no. 1086 R1 - Unit 1 Cycle 27 Core Reload Modification

Description of Change

This activity will replace depleted fuel from the Unit 1 Cycle 26 reactor core with 41 feed (fresh) fuel assemblies. This will allow the Unit 1 reactor to produce power at its rated capacity in Unit 1 Cycle 27 for approximately 18 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. Revision 0 of this evaluation is valid only for Modes 3, 4, 5 and 6.

Revision 1 of this evaluation is valid for operation of Unit 1 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 1 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 ZIRLO™ 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 1 Cycle 27 is consistent with Prairie Island's Current Licensing Basis.

50.59 Evaluation no. 1088 R0 - Auxiliary Feedwater Increased Flow to SG due to Recirculation Line Flow Meter Install

Description of Change

The proposed change will install an orifice plate flow meter in Unit 1 Turbine Driven Auxiliary Feedwater Pump recirculation line downstream of the existing breakdown orifice. The installation will include a 1" orifice plate meter body assembly, flow gauge, and associated tubing, valves and hardware.

The new flow meters in the Auxiliary Feed pump recirculation lines will provide flow indication where it does not presently exist. The flow indication will allow the recirculation lines to remain in service during pump testing which in turn will keep the pump lube oil coolers in service. The result is the pump and turbine bearing temperature will not be challenged during testing. This 50.59 evaluation will specifically review the increased flow to the steam generators caused by change.

Summary of 50.59 Evaluation

The increase flow to the steam generators as a result of the addition of the flow transmitter orifice in the 11 Turbine Driven Auxiliary Feedwater pump recirculation line, remains bounded by the input assumptions used in the margin to overfill analysis during a Steam Generator Tube Rupture and in the Main Steam Line Break analyses. Therefore there is no change to the consequences of an accident, malfunctions or impact on a design basis limit for a fission product barrier. The Auxiliary Feedwater System is not an initiator of an accident or so there is no new accidents created and the frequency of accidents and malfunctions has not been increased. Hence NRC approval prior to implementation of the installation is not required.

50.59 Evaluation no. 1089 R0 - Incorporate New Analysis of Record for the Units 1 and 2 Loss of Load/Turbine Trip Peak RCS Pressure Event

Description of Change

The activity discussed in this evaluation is the incorporation into the Current Licensing Basis a new Analysis of Record for Prairie Island Unit 1 and Unit 2 Loss of Load/Turbine Trip Peak Reactor Coolant System Pressure Event. The activity is required due to a change in the pressurizer pressure uncertainty input to 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 pressurizer pressure uncertainty, used an NRC approved methodology and the results showed that the design limits as currently described in the Prairie Island UFSAR were 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.

ENCLOSURE 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
CHANGES TO REGULATORY COMMITMENTS

Regulatory Commitment Change 10-01 – Delete Commitment to install non-intrusive RCS level indication system on Unit 1. This indication will only monitor level in the diameter of the loop piping and not the total RCS level.

Initial commitment was made via letter to the NRC, "Reply to a Notice of Violation, NRC Inspection Report No. 306/92006, Inadequate Procedure for Draindown to Midloop", dated June 15, 1992.

The non-intrusive level indicator was credited as independent level instrumentation because, at that time, the other level indications were pressure compensated from the same pressure transmitter and could not be credited as independent systems.

Based on plant and procedure changes, the plant is currently configured such that A and B Train ERCS D/P Level indicators no longer require pressure compensation when the RCS is vented and the Refueling Canal Level Instrument is also available for monitoring level. Therefore, without reliance on the Ultrasonic level indicators, the station has two independent, continuous RCS water level indications available for reduced inventory conditions, which meets the requirements of GL 88-17.

Regulatory Commitment Change 10-02 – Delete Commitment to install non-intrusive RCS level indication system on Unit 2. This indication will only monitor level in the diameter of the loop piping and not the total RCS level.

Initial commitment was made via letter to the NRC, "Reply to a Notice of Violation, NRC Inspection Report No. 306/92006, Inadequate Procedure for Draindown to Midloop", dated June 15, 1992.

The non-intrusive level indicator was credited as independent level instrumentation because, at that time, the other level indications were pressure compensated from the same pressure transmitter and could not be credited as independent systems.

Based on plant and procedure changes, the plant is currently configured such that A and B Train ERCS D/P Level indicators no longer require pressure compensation when the RCS is vented and the Refueling Canal Level Instrument is also available for monitoring level. Therefore, without reliance on the Ultrasonic level indicators, the station has two independent, continuous RCS water level indications available for reduced inventory conditions, which meets the requirements of GL 88-17.

Regulatory Commitment Change 10-03 – Delete Commitment for Working Hour Restrictions

Original commitment to establish policies for working hour restrictions was made via letter to the NRC, "Additional Information Related to Implementation of NRC Guidelines for Working Hours", dated November 16, 1982

The working hour restrictions have now been codified through 10 CFR Part 26, Subpart I (73 FR 16966, March 31, 2008), and the above referenced commitment can be cancelled.

Regulatory Commitment Change 11-01 – Change wording for the Requirement for RCS Intact

Original commitment to create new procedures for reduced inventory operations with the Reactor Coolant System intact was made via letter, "Reply to Notice of Violation, NRC Inspection Report No. 306/92006, Inadequate Procedure for Draindown to Midloop", dated June 15, 1992.

The phrase "with the Reactor Coolant System intact" has been removed. The use of the phrase "Reactor Coolant System intact" within the sentence is confusing and could be inferred to mean the RCS must be intact for reduced inventory operations. This commitment change does not involve changes to the hardware modifications or the procedure actions for use of the hardware modifications. Additionally, the change in commitment wording does not alter how the plant performs draining activities to reduced inventory or any of the hardware installed to perform reduced inventory operations.