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U S Nuclear Regulatory Commission
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60
PRAIRIE ISLAND NUCLEAR GENERATING PLANT SAFETY EVALUATION
SUMMARY REPORT

With this letter, the Nuclear Management Company, LLC, (NMC) submits two attachments. Attachment 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.

Attachment 2 contains discussion of changes to regulatory commitments made within our Regulatory Commitment Change Process.

In this letter we have made no new Nuclear Regulatory Commission commitments. Please contact Jeff Kivi (651-388-1121) if you have any questions related to this letter.

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Attachments

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ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282. 50-306

December 10, 2003

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS, AND EXPERIMENTS – JULY 2003

49 pages follow

Below are a brief description and a summary of the safety evaluation for each of those changes, tests, and experiments which were carried out without prior Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(b).

Non-Modification Safety Evaluation 285 (Addendum 2) – Removing Loop A Cooling Water Return Header from Service

Description of Change

Unit 1 emergency diesel generator, D1, cooling water return isolation valve, CW-62-2, failed closed due to the stem separating from the disc. Due to the design of this system it was necessary to isolate and drain the return header to perform repairs. This Safety Evaluation was performed to evaluate removing Loop A Cooling Water Return Header from service, draining (for the purpose of repairing CW-62-2), refilling, and returning to service.

Summary of Safety Evaluation

Technical Specifications allow removal of a Cooling Water (CL) Loop from service. Removal of CL Loop A from service also requires removal from service of safeguards equipment supplied by this loop. Since this loop provides cooling to the Train A Component Cooling (CC) Water System, the safeguards equipment supplied by the Train A CC system also is required to be removed from service. The CL dump to grade flow path would have to be used to drain the portion of the Loop A CL Return Header in a timely manner. Use of this flow path for CL Loop A isolates the CL Loop B dump to grade path. Normal CL Loop B return flow path would continue to be operable and since these activities are under the Technical Specifications, a second single or active failure is not required to be assumed. The safety evaluation concluded that Loop A Cooling Water Return Header could be removed from service in accordance with the proposed plan since the plant Technical Specifications allows the activities and CL Loop B remains in service.

Non-Modification Safety Evaluation 321 (Rev. 2) – Safeguards Chilled Water System Seismic Adequacy

Description of Change

This evaluation demonstrated the seismic adequacy of the Safeguards Chilled Water System to maintain pressure boundary integrity during a seismic event. Revision 2 to this safety evaluation contains the plant walkdown information committed to in the previous revision to this safety evaluation. This evaluation of the system did not include 121 and

122 Control Room Chillers; which are being evaluated as part of the Unresolved Safety Issue (USI) A-46 seismic review.

Summary of Safety Evaluation

During the original design, the piping in the Safeguards Chilled Water System (SCWS) was not specifically analyzed for dynamic seismic loadings. In lieu of performing these analyses, a deliberate engineering effort was made in justifying the lack of a dynamic analysis for the SCWS. As part of this engineering effort, system walkdowns were performed. All piping and supports and associated equipment of the safeguard chilled water system were found to be seismically adequate for the Prairie Island design basis earthquake, with the exception of the 121 and 122 Control Room Chillers. Based on the design and construction of the Design Class III SCWS piping, a comparison to demonstrated piping capabilities during seismic events at other facilities, and the seismic adequacy review, it is concluded that this piping would perform successfully during a seismic event. That is, the piping would maintain pressure boundary integrity.

Non-Modification Safety Evaluation 477 – Opening of Selected CC and CL Breakers for Appendix R Concerns

Description of Change

The purpose of this safety evaluation is to justify maintaining certain motor control center (MCC) breakers open during normal operation. Maintaining these breakers open eliminates concerns associated with spurious operation in the event of a fire requiring Appendix R shutdown. Based on the requirements of 10 CFR 50, Appendix R and NRC Generic Letter 86-10, in the event of an exposure fire, spurious operation is required to be assumed for equipment that could be affected but not required to be operable.

Summary of Safety Evaluation

The Safety Evaluation concluded that this change would not result in an unreviewed safety question, nor would it affect or change Plant Technical Specifications. These changes provide compliance with the requirements of 10CFR50 Appendix R.

Non-Modification Safety Evaluation 483 – Appendix R Safe Shutdown Analysis Implementation

Description of Change

The purpose of this safety evaluation is to implement the updated analysis of the Prairie Island system relative to the fire protection safe shutdown requirements of 10CFR50 Appendix R. The need for a re-analysis resulted from:

1. Requirements to resolve issues associated with the qualification of Thermo Lag as an approved fire barrier material: the economics of installing an alternate material demanded that an analysis be completed to minimize the amount of fire barrier material required to protect safe shutdown equipment.
2. Completion of the Prairie Island Nuclear Generating Plant (PINGP) Station Blackout/Electrical Systems Upgrade projects. The Station Blackout project installed two, redundant Unit 2 emergency diesel generators (D5 and D6). The Electrical Systems Upgrade Project made extensive design modifications to both the Unit 1 and Unit 2 Safeguards Electrical distribution systems.
3. The need to reconstitute detailed analysis background documentation.

Summary of Safety Evaluation

The Safety Evaluation concluded that this change would not result in an unreviewed safety question, nor would it affect or change Plant Technical Specifications. These changes provide compliance with the requirements of 10CFR50 Appendix R.

Non-Modification Safety Evaluation 505 – Control Room Ventilation Zone Door Seal Substitution

Description of Change

This evaluation changes Updated Safety Analysis Report (USAR) Figure I.14-7 for the typical door configuration for the control room, relay room and other sensitive areas from one requiring closed cell sponge neoprene seals to a more generic statement requiring closed cell sponge neoprene or equal.

Summary of Safety Evaluation

The USAR requires the Control Room, Relay Room, and any other sensitive areas to be provided with seals on doors and penetrations in order to preclude adverse environment following a high energy line rupture or crack. The USAR figure, which shows that the control room, relay and other sensitive area doors are sealed, specifically identifies sponge neoprene as adequate protection for the control room in the event of a steam line break. Sponge neoprene is not the only material available to provide adequate sealing of these doors. Changing the USAR allows changing to better quality seals than neoprene. Two specific gasket materials, 1) a magnetic seal consisting of a ferritic material enclosed in a neoprene rubber and chlorine covering, and 2) a silicon rubber gasketing material, were evaluated for their adequacy to seal these doors. The evaluation concluded these seals are acceptable and can be considered equal to the seal shown in USAR Figure I.14-7.

Non-Modification Safety Evaluation 506 – Operations Manual, Section F5, Appendix E, Rev. 4 Changes and Calculation GEN-PI-026, Rev. 2 Changes

Description of Change

The purpose of this safety evaluation is to evaluate the proposed changes to Operations Manual, Section F5, Appendix E, "Fire Protection Safe Shutdown Analysis Summary." Revision 4. The changes, in general, involve:

1. Adding information to address the proposed USAR changes which include (a) a note to indicate that F5, Appendix E, is part of the USAR; (b) a summary of the III.G safe shutdown analysis results in the Executive Summary Section 1.2; and (c) a discussion the alternate shutdown methodology in a new section 2.7.3.

2. Replacing Section 2.3.6.A paragraph on offsite power to reflect the revised safe shutdown analysis, which credits emergency diesel generators D2 and D6 and updating Table 1 Safe Shutdown Equipment Summary.

3. Relocating Appendix R Section III. G criteria from the Executive Summary Section 1.2 to the assumptions Section 2.1 (Note: this information was not necessary in the Executive Summary).

4. Revising Section 5.0 to clarify that installation of emergency lighting was based on operator actions for any fire, not just a Control Room/Relay Room fire, and to clarify the periodic surveillance method for the emergency lights.

5. Revising the Table of Contents to include Section 3.7, which was erroneously omitted from Revision 4.

6. Table 2 was revised to insert safety injection (SI) accumulator valves MV-32174 and MV-32175 in the list of breakers that are maintained open to prevent spurious operation. These valves were missed in the Revision 3 of F5, Appendix E that was evaluated in SE-483. Table 2 was also revised to remove the "*" designation for valves MV-32200 and MV-32211 because the breakers are actually credited for Appendix R which were also inadvertently missed during Rev. 3 of F5, Appendix E (SE-477).

This evaluation also evaluated the changes made to Revision 2 of the Appendix R safe shutdown analysis (SSA). The calculation revision supports the changes described in item 2 for F5, Appendix E.

Summary of Safety Evaluation

The relocation of information from Section 1.2 to Section 2.1 (Item 3) and the inadvertent omission of Section 3.7 from the Table of Contents (Item 5) are editorial and do not change the information currently contained in F5, Appendix E. Changes in Item 6

incorporate the current Revision 1 of the SSA and updates Table 2 to be current with the calculation. Therefore, Items 3 and 5 are not safety significant and were not evaluated. Item 6 changes have been previously evaluated in Safety Evaluation Nos. 477 and 483. The Safety Evaluation concluded that the change for the remaining items (1, 2, and 4) would not result in an unreviewed safety question, nor would they affect or change Plant Technical Specifications. These changes provide compliance with the requirements of 10CFR50 Appendix R.

Non-Modification Safety Evaluation 575 (Rev. 1) - Modification 02CL02

Description of Change

Prior to the installation of T-Mod 00T077, the 12 and 22 safeguards cooling water pumps used non-safety related components to provide the water source to the pump shaft bearings. It had been determined that a safety related source of water using safety related components should be provided. T-Mod 00T077 installed this safety related bearing water supply and was evaluated under SE575. This new revision to this evaluation adds reference to the permanent modification with no changes made from the T-mod.

Summary of Safety Evaluation

This modification increases the reliability of 12 and 22 safeguards cooling water pumps by providing a safety related source of water to the shaft bearings. This design is independent of the supplies to the other safeguards pump, it is Quality Assurance Class 1C and Design Class 1. Evaluations of the potential effect on other components show that this modification is acceptable. This modification does not increase any consequences, does not affect any accident initiating sequences, does not increase the probability of any evaluated malfunctions or create any new types of malfunctions and does not decrease any margins of safety. Therefore, there are no unreviewed safety questions.

Non-Modification Safety Evaluation 575 (Addendum 1, Rev. 1) - Modification 02CL02

Description of Change

Prior to the installation of T-Mod 00T078, the 121 Motor Driven Cooling Water Pump (MDCLP) used non-safety related components to provide the water source to the pump shaft bearings. It had been determined that a safety related source of water using safety related components should be provided. T-Mod 00T078 installed this safety related bearing water supply and was evaluated under SE575, Addendum 1. This new revision to this SE addendum adds reference to the permanent modification with no changes made from the T-mod.

Summary of Safety Evaluation

This modification increases the reliability of 121 MDCLP by providing a safety related source of water to the shaft bearings. This design is independent of the supplies to the other two safeguards pumps, it is Quality Assurance Class 1C and Design Class 1. Evaluations of the potential effect on other components show that this modification is acceptable. This modification does not increase any consequences, does not affect any accident initiating sequences, does not increase the probability of any evaluated malfunctions or create any new types of malfunctions and does not decrease any margins of safety. Therefore, there are no unreviewed safety questions.

Non-Modification Safety Evaluation 584 – Appendix R Document Update - Containment Sump B Valve Hot Short Issue Resolution

Description of Change

The purpose of this Safety Evaluation is to evaluate proposed changes to the Appendix R Program to reflect plant changes made under design changes 99SI02 (Containment Sump B Valve Hot Short Issue) and 00FP01 (Kaowool Replacement) to mitigate the issues with the Containment Sump 'B' to residual heat removal (RHR) motor valves identified in Prairie Island Licensee Event Report (LER) 1-98-15.

Summary of Safety Evaluation

The evaluation concluded the changes to the Appendix R program required to implement the fixes installed under design changes 99SI02 and 00FP01 to mitigate the Containment Sump B to RHR Motor Valve hot short issue can be incorporated without adversely affecting the ability of the plant to achieve and maintain hot shutdown. The evaluation also concluded that this change would not result in an unreviewed safety question, nor would it affect or change Plant Technical Specifications. These changes provide compliance with the requirements of 10CFR50 Appendix R.

Non-Modification 50.59 Evaluation 1000 – Containment Pressure Response to Changes to Table of Structural Heat Sinks

Description of Change

As a result of a detailed review, the table of structural heat sinks used in the containment integrity analyses is being changed. This change includes performing new containment analyses for both the loss of coolant accident (LOCA) and main steam line break (MSLB) and making associated updates to the USAR.

Summary of 50.59 Evaluation

The table of structural heat sinks and the associated containment integrity analyses evaluate post-accident response and are not accident initiators; thus, there is no increase in frequency of any accident or creation of an accident of a different type. Approved methods of evaluation are used for the containment integrity analyses performed to evaluate these changes. The results from these analyses show that the containment peak pressure is maintained below the design basis limit of 46 psig; therefore, a design basis limit for a fission product barrier (DBLFPB) is not exceeded or altered. The predicted containment pressure profile is less than that assumed for determining the containment leakage rate in the dose analyses. Thus, there is no change to the consequences of any accident analyses or due to any equipment malfunctions. The results from the containment integrity analyses and any other affected analyses were reviewed to ensure that equipment important to safety would not be adversely affected. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

Non-Modification 50.59 Evaluation 1004 – Spent Fuel Pool Cooling

Description of Change

This 50.59 evaluation has two primary objectives:

1. Update the USAR with revised Spent Fuel Pool (SFP) cooling thermal-hydraulic analyses following the replacement of 122 SFP Cooling Heat Exchanger.

2. Provide a basis for the time required to provide a makeup water supply to the pool(s) in the event of a loss of cooling and increase this time period from ten minutes to one hour.

Summary of Safety Evaluation

The updated SFP cooling analyses and make-up time requirements are not accident initiators; thus, there is no increase in frequency of any accident nor creation of an accident of a different type. The methods of evaluation used for determining the pool heat load is the same as that discussed in the USAR. The results from these analyses show that adequate cooling of the spent fuel is maintained; therefore, a DBLFPB is not exceeded or altered. As a Fuel Handling Accident is not initiated by these changes, there is no change to the consequences of any accident analyses or due to any equipment malfunctions. The results from the cooling analyses show that equipment important to safety would not be adversely affected. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

Non-Modification 50.59 Evaluation 1005 – Motor-Operated Valves 32335 and 32336 Anti-Hammering Wiring Change

Description of Change

A change in wiring is proposed for Motor Valves MV-32335 and MV-32336 to prevent the possibility of motor valve damage due to repeated valve motor operation (valve hammering). The proposed fix for the hammering concern is to rewire the motor valve such that an additional 33/ac contact is placed in series with the motor valve 'close' control switch contact in the valve closing series. The hammering concern and fix are described in detail on the attached sheet.

Summary of Safety Evaluation

The change to MV-32335 and MV-32336 will not change the physical properties of the valve. The changes will be in valve actuator operational logic. The valve will operate the same as in the past under all conditions except one. This would be the highly unlikely scenario that the valve would be between 70 and 100% closed and a loss of power to the valve would occur. When power is restored the valve would have to be given an open signal to "reset" the anti-hammering logic before the valve could be given a closed signal. The valve would still be able to be positioned either electrically or manually the same way as any other valve in the plant would be.

This along with the replacement cabling and additional contacts do not represent a change that would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an system, structure or component (SSC) important to safety.

Non-Modification 50.59 Evaluation 1006 - Shutdown without 2RX Reserve Transformer

Description of Change

The 2RX transformer and the 2M transformer are the normal power supplies to 4KV buses 21 and 22. Buses 21 and 22 power the unit 2 Reactor Coolant Pumps (RCPs) and Main Feedwater Pumps (FWPs). During normal power operation, unless it is not available, the 2M transformer provides the power to these buses. During shutdown operation, the normal power supply is from the 2RX transformer. During the shutdown procedure, the power to buses 21 and 22 is transferred from 2M to 2RX so that when the turbine is taken off line, these buses are still powered. On November 8, 2001, the 2RX transformer locked out, due to tripping of the 51G relay. Without the 2RX transformer available, then buses 21 and 22 would not have an alternate power supply during a normal shutdown. Per C20.3 AOP4, with a loss of 2RX, buses 21 and 22 would be powered from 1R transformer. However, with the 1M transformer out of service, there are loading concerns for the 1R transformer. The purpose of this change to shutdown procedure 2C1.3 is to provide instructions for shutting down the plant

without the 2RX transformer available. In the current plant configuration, this would necessitate a manual reactor trip with a loss of forced reactor coolant flow and normal feedwater flow.

Summary of Safety Evaluation

The change to the procedure will not have any effect on the frequency with which the plant is shutdown. For an unplanned shutdown, another unrelated failure, requiring a plant shutdown, would have to occur. If the reactor were to trip (manually or automatically) in response to a malfunction of some type, then E-0 is used. The scenario is bounded by transients already analyzed, which show that all acceptance criteria are satisfied. Therefore, a DBLFPB is not exceeded or altered. There is no change to the consequences of any accident analyses or due to any equipment malfunctions. The conditions are within the design capabilities of the structures, systems and components; thus, equipment important to safety would not be adversely affected. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

Non-Modification 50.59 Evaluation 1007 – Steam Generator Level and Containment Temperature Restrictions in support of MSLB Analysis

Description of Change

The Nuclear Analysis Department (NAD) recently discovered that their MSLB containment integrity analysis methods topical report considered the failure to close of a steam generator (SG) non-return check valve. Although it is uncertain that it is a requirement to assume this as an active single failure, an analysis was run with this conservative assumption. The purpose of this evaluation is to justify the acceptability of imposing new restrictions on SG narrow range level and containment temperature in support of the MSLB containment integrity analysis (this will simply be referred to as the MSLB analysis in this evaluation). These changes are considered acceptable because they will not adversely affect plant operation and will ensure that the peak containment pressure during a MSLB will not exceed 46 psig.

Summary of 50.59 Evaluation

The changes made to input parameters made in the MSLB containment integrity analyses are for post-accident response and are not accident initiators; thus, there is no increase in frequency of any accident or creation of an accident of a different type. Approved methods of evaluation are used for the containment integrity analyses performed to evaluate these changes. The results from these analyses show that the containment peak pressure is maintained below the design basis limit of 46 psig; therefore, a DBLFPB is not exceeded or altered. Containment integrity is maintained; thus, the dose analysis for the MSLB inside of containment is still bounded by the MSLB outside of containment. Thus, there is no change to the consequences of any accident analyses or equipment

malfunction. The results from the containment integrity analyses and any other affected analyses were reviewed to ensure that equipment important to safety would not be adversely affected. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

Modification 96AC01 (Rev. 1) – Nuclear Steam Supply System Annunciator System Upgrade

Description of Change

This modification upgraded the nuclear steam supply system (NSSS) annunciator system to a microprocessor-based system similar to the BOP (balance of plant) annunciator system. The BOP annunciator system was upgraded in 1990 and 1991. The new NSSS system improves both reliability and flexibility of the NSSS annunciator system.

Summary of Safety Evaluation

- The modification did not alter the radiological consequences of any accident in the USAR. It did not affect any of the inputs, assumptions, or methods in the dose analyses and did not affect any fission product barriers. The NSSS annunciator system is not required for post-accident monitoring.
- The modification affected only the non-safety related NSSS annunciator system. This system does not initiate any accidents.
- This modification did not change the functional or operational requirements of the NSSS annunciator system. The system does not perform any functions required to mitigate an accident, nor does it affect any system required to mitigate an accident.
- This modification did not change the radiological consequences of any malfunction of safety-related equipment.
- The USAR does not evaluate a malfunction of the NSSS annunciator system.
- There are eight Technical Specification related NSSS annunciators – none of which is required to be operable at cold shutdown.

Modification 96AF01 (Part 2, Rev. 1) – Auxiliary Feedwater Pump Runout Protection

Description of Change

The auxiliary feedwater pumps have low suction pressure and low discharge pressure switches that will stop the pump if low pressure occurs. The low suction pressure switch

protects the pump from a loss of suction and the low discharge pressure switch protects the pump from runout conditions. The purpose of the modification is to provide a bypass of the low discharge pressure trip for the turbine-driven auxiliary feedwater pumps during ATWS (anticipated transient without scram) conditions by blocking the low discharge pressure trip when the reactor trip breakers are closed. This circuit ensures that following completion of the AMSAC (ATWS mitigating sensing actuation circuitry) initiation of auxiliary feedwater (AFW) during an ATWS, the turbine-driven auxiliary feedwater pump (TDAFWP) will continue to run.

Part 2 of this modification separated the time delay circuit for the low discharge pressure and low suction pressure switches on the 11 and 22 TDAFWPs. The existing time delay relay remains in place for only the low suction pressure switch (with the same time setting). A time delay relay was added for the low discharge pressure switch (with a new time setting). Contacts from the 11 and 22 TDAFWP control switch Auto and Shutdown Auto modes were designed into the circuit. Also, a RTA relay contact for 11 TDAFWP (RTB relay contact for 22 TDAFWP) is used to detect reactor trip breaker position for the low discharge pressure trip circuit.

Summary of Safety Evaluation

The Safety Evaluation determined that the modification did not represent an unreviewed safety question because:

- The modification ensures the operability of the pressure trip circuitry and the operation of the TDAFWPs during an ATWS.
- The new relays are identical to the original equipment, so the new relays do not introduce new failure modes.
- During ATWS, the TDAFWPs are allowed to operate without low discharge pressure trip, since the events causing low steam generator pressure will not cause runout condition.

Modification 96EB01- 480V Common Loads Conversion

Description of Change

This modification added the capability of operating 480V common safeguards loads from power sources derived from either Unit. There are cases now where discrete administrative controls are necessary to operate equipment that is common to both Units but is presently fed from Unit 1. In order to maintain the 480V and 4160V sources to these shared loads, administrative controls are necessary when unit 1 voltage sources are in "maintenance". This usually occurs during refueling outages. Manual transfer switches, sized for the connected load, were installed for MCC's that serve

common loads. These transfer switches are now the controlling point in order to feed downstream MCC's from Unit 1 or 2. The practice of using transferable MCC's to feed shared loads has already been implemented at Prairie Island for the MCC's.

Additionally, certain loads fed from MCC's other than ones connected to the transferable MCC's have been relocated from their existing MCC to the MCC's that are transferable. MCC's 1MA1 and 1MA2 are the ones selected for being fed from either unit through their respective transfer switches. MCC 1MA1 previously did not exist. This MCC was eliminated administratively during the interface portion of the Station Blackout/Electrical Safeguards Upgrade (SBO/ESU) project by connecting its bussing with MCC 1M1. MCC 1M1 and old 1MA1 are adjacent to each other. Each MCC is comprised of a combination of three stacks of motor starters and breaker enclosures. The split between MCC 1M1 and revised 1MA1 MCC was performed by removing the interconnecting cabling between the bussing. An additional power cable was installed from the transfer switch to MCC 1MA1. This cable was an existing plant abandoned cable and resurrected for this purpose. Typical 1MA1 and 1MA2 loads are aux building fan loads, filters and filter heaters. Typical loads for MCC 1T1 and 1T2 are control room lighting, radiation monitors, relay room cooling fan power and event monitoring room cooling fan power. The unit cooler fan motors are 230V, 1 phase, 60Hz, however and a calculation has been performed for a new transformer and panel board to demonstrate there is sufficient current carrying capacity. All four boric acid heat trace switchrack transformers were transferred from their existing Unit 1 MCC's to MCC 1T1 and 1T2.

There are also unitized loads, which are presently fed from MCC's that have the capability of being fed from either unit that will be relocated to unitized MCC's. These typically would be steam supply valves to AFW Pumps, all four hydrogen recombiners and shield building ventilation equipment.

The SFP cooling pumps were powered from a normal building source. The 480V common loads project converted these two pumps and their controls from MCC 1GA1 and 1GA2 to 1T1 and 1T2 respectively. This power source change is diesel backed and allows the SFP pumps greater flexibility in availability. These pump power sources are typically bus tied at the Bus190/290 level during refueling outages, offering only one power source to two pumps. The new configuration provides at least one power source to each spent fuel pit pump. The circuit breaker for the SFP pumps within the MCC is being used as the isolation device between Class 1E and Class III. The control circuit is upgraded to Class 1E safety-related to satisfy the single failure criteria expressed in the Prairie Island Engineering Manual. Although these two pumps are now fed from a safeguards power source, the pumps will not be reclassified and not tested under an existing pump In-service Testing (IST) program. The 121/122 SFP pumps will remain non-safety related devices.

Summary of 50.59 Evaluation

All relocated loads are fed from a safeguards power source, with cables routed in an analyzed environment. No changes in function were introduced for the equipment used

in accident analysis, which are affected by this modification. No changes in load sequence step were made by this modification. The changes in electrical loading were all within the applicable equipment ratings and load limits. The introduction of transfer switches for the MA MCC's is consistent with previous plant design and equipment installed to serve similar functions for the AB MCC's. The diesel generator and other electrical loading acceptance criteria as stated in the USAR are not exceeded by this modification.

Thus the response to each of the seven questions was "No" concluding that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 96MP04 - Generic Piping Replacement Modification

Description of Change

Periodically, there is a need to replace piping components due to service related wear or due to fouling of the piping systems. Many design enhancements can be incorporated at relatively low additional cost during these replacements. These enhancements to the systems improve maintainability of the systems, allow for inspections to monitor for degradation and at times eliminate the degradation mechanism. This Design Change will allow piping replacement and associated piping component changes on non-safety related, small bore[1], moderate energy piping[2]. Additionally, if changes to the pipe routing or pipe size are made or components are added or removed, then the plant design change is further limited to piping systems that were designed with ASME B31.1 "cookbook" methods, rather than those that were designed with a dynamic piping stress analysis. It is anticipated that flanges may be added, valves may be added or that valves may be removed within the scope of this Design Change.

Piping that is subject to the guidance of NRC Regulatory Guide 1.143, "Design Guidance of Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Cooled Nuclear Power Plants," is outside the scope of this Design Change.

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[1] For the purposes of this modification, small bore piping is defined as piping 3 inches in diameter and smaller.
[2] Moderate energy piping is defined as piping that, during normal plant conditions, is either in operation or maintained pressurized (above atmospheric temperature) under conditions where both of the following are met:

a. Maximum operating temperature is 200°F or less, and

b. Maximum operating pressure is 275 psig or less.

Summary of Safety Evaluation

During installation of piping within the scope of this Design Change, there may be a need for revisions to the USAR, which would require a safety evaluation for simple changes like pipe size changes or installation of isolation valves in non-safety related systems when revising the associated flow drawings. This safety evaluation provides the justification for changes to flow drawings (and ultimately the USAR) for non-safety related systems.

Installation of replacement piping components restores the plant to new or improved conditions. Non-safety related equipment is not relied upon to mitigate accidents. Design margins are in line with those assumed in the USAR to assure malfunctions do not occur. Replacement and rerouting of small bore lines restores system piping margins to Code allowable levels. All replacements are in non-safety related piping, yet are performed to the Code to ensure the reliability of equipment important to safety. Appropriate engineering factors are incorporated within the design to preclude failure. Thus, the evaluation concluded this modification did not require NRC review and approval and did not affect or change plant Technical Specifications.

Modification 96SA01 – 121 and 122 Instrument Air Dryer Isolation Valves

Description of Change

The loss of instrument air header pressure through a malfunction of the existing purge exhaust valves for the instrument air dryers has caused a unit trip in the past due to the feedwater regulating valves closing on low instrument air pressure (see LER 96-02). This modification provided added protection against the loss of instrument air header pressure due to a failure of the purge exhaust valves by installing an automatic instrument air dryer purge exhaust isolation valve, which closes upon decreasing instrument air header pressure. The instrument air system is needed for normal and abnormal operations of the plant and to recover from an accident. However, in general, the instrument air system is classified as non-safety related with the exception of a few backup air accumulator systems, which are required to operate or maintain safety related equipment in a safe condition. This modification will be on non-safety related portions of the instrument air System.

Summary of Safety Evaluation

The Safety Evaluation concluded that there would be no adverse impact to the USAR, Technical Specifications or design basis of the plant and no unreviewed safety question exists as a result of this modification.

Modification 97CL04 – 121 MDCLP Pressure Switch Separation

Description of Change

The 121 Motor Driven Cooling Water Pump (MDCLP) was upgraded to Safeguards status under the SBO/ESU modification 89Y976. This pump can be powered from either a Train A or Train B Bus, depending on if 12 Diesel Driven Cooling Water Pump (DDCLP) or 22 DDCLP is out of service.

The 121 MDCLP will automatically start on a cooling water low header pressure signal from pressure switch PS-16259. This pressure switch can be considered either a Train A or Train B instrument depending on the alignment of 121 MDCLP. PS-16259 is mounted on the same panel as the pressure switch PS-16002 for 12 DDCLP (Train A). The sensing lines for these pressure switches are routed together for a short distance. The cables to the pressure switches are together at the mounting panel.

Redundant instruments and sensing lines used with Safeguard systems must be separated by a minimum of 36 inches free air space between redundant instruments. Cables in conduits are to be separated by a minimum of 1 inch. Thus, when 121 MDCLP is aligned as the Train B safeguards Cooling Water pump, adequate separation is not provided for the pressure switches.

Pressure switch 16259 was reclassified as safety related. In order to provide required separation between instruments, sensing lines and cables, this modification moved PS-16259 to a new location and upgraded the associated cable to Safety-Related. It also moved the pressure sensing line for PS-16002. This modification resolved the switch separation issues identified in LER 1-97-01 dated 10-24-97.

Summary of Safety Evaluation

The safety evaluation concluded that this modification would not result in an unreviewed safety question and would not affect or change Plant Technical Specifications.

Modification 97EB01 - Anchorage for Bottom Feed Motor Control Centers

Description of Change

This change added seismic restraint brackets at the base of four MCC's. The design provided for four 6-inch by 6-inch angles that were attached between the MCC base frame and the floor at each MCC. One end of the angle was bolted to the MCC frame while the other end was welded to a baseplate and anchor bolted to the floor. Also, additional bolting was included to secure the bottom base of the MCC to the MCC mounting channel frame at each of the MCC's four corners.

The installation of the new seismic restraints did not affect the normal operation of the MCC's. The new restraints are passive components and would only affect the MCC's during a PINGP design bases seismic event. The design intent of the new restraints is to improve the capability of the MCC to perform during and/or after the postulated PINGP seismic event. Therefore, there is no adverse impact on safety during the operation phase or operation post accident.

Summary of 50.59 Evaluation

The evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 97FP26 (Rev. 0, 1, and 2) - Diesel Generator Source Breaker Modification

Rev. 0

Description of Change
This modification added a Ready to Load Relay (RTLR) contact into the breaker closing circuits for the D1 (Unit 1) and D5 (Unit 2) Emergency Diesel Generators, breakers 15-2 and 25-2, respectively. The addition of this contact prevents the breakers from being closed until the diesels are up to speed and ready to accept load. The relays are safety-related relays with an existing contact currently utilized as an input to the load sequencer for each diesel. These relays are tested regularly and the addition of a second contact from these relays will not significantly affect the overall reliability of the diesel generators or their source breakers.

For breaker 15-2, a normally closed (NC) contact from the RTLR was converted to a normally open (NO) contact and wired in series with the lockout relay contacts in the breaker closing circuit. For Breaker 25-2, an existing NO contact from the RTLR from the local closing circuit was relocated so that it will be in the circuit or all of the closing schemes (i.e. auto, local manual and remote manual).

Rev. 1

Description of Change

The D5 Diesel Generator close circuit was modified per Revision 0 of this Design change described above. The D5 Source Breaker portion of this design change was successfully installed, pre-op tested and turned over. The monthly testing was performed and an error message occurred since the close circuit now had an open contact that wasn't bypassed by the load sequencer.

This design change (Rev. 1) revised the earlier change and rewired the RTLR contact to a point above where the closing circuit from the load sequencer enters the common

closing scheme. This provides the same amount of protection for a relay/control room fire as the earlier change provided. However this location will allow the D1 (D5) load sequencer to check the closing circuit for breaker 15-2 (25-2) without installing a jumper around the RTLR contact.

Rev. 2

Description of Change

This design change still added a RTLR contact into the breaker closing circuit for the D1 and D5 breakers. The addition of this contact will prevent the breakers from being spuriously closed as a result of a Control Room fire until the diesels are up to speed and ready to accept load. The relays are safety-related relays with an existing contact currently utilized as an input to the load sequencer for each diesel.

Summary of Safety Evaluation (Rev. 0, Rev. 1, and Rev.2)

D1 and D5 are not accident initiators. The design change only changes the close scheme of the D1 and D5 diesel generators in a place that does not affect their operation for any design basis accident, malfunction, or event. This design change limits fire damage to equipment and systems required by the Appendix R Safe Shutdown Analysis to achieve and maintain safe shutdown conditions. Thus, the evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 97ZH01 – Provide Limit Stops for Control Room Chiller

Description of Change

This design change added mechanical limit stops around the equipment mounting spring isolators for the 121 and 122 Control Room Chillers and the D1 and D2 diesel generator control panels. These stops limit the potential for lateral movement during a design basis earthquake.

This design change is one of the actions taken in response to USI A-46.

Summary of Safety Evaluation

This design change has no potential to increase on-site or off-site dose. This design change increased the reliability of the affected equipment during a design basis earthquake (DBE) and did not affect the components performance during normal operation or in response to a design basis accident (DBA). The design change decreases the probability of equipment malfunction during a DBE and does not increase the probability of malfunction during a DBA. The evaluation concluded that this

modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 97ZS01 - Shield Building Vent System (SBVS) Indicating Lights Separation

Description of Change

Both trains of Units 1 & 2 Shield Building (Annulus) Vent Exhaust Fan Motor Dampers have indicating light cables running through the same cable tray. The section of cable tray where this occurs is non-trained tray (black cables). These cables feed the Control Board status lights on "A" panel. These cables also share the same terminal box as the motor damper control cables, which are trained cables. There is no isolation between the indicating light cables and the damper control cables.

To improve on the design of this system, this modification installed fuses for isolation of the status light portion of the control cabling. This effectively isolates the trained portion of the circuit (motor damper) from the non-trained indicating light circuit.

Summary of 50.59 Evaluation

There is no impact on the USAR or the ISFSI SAR for this design.

Sections 8.7.2, "Cable Tray Routing", and 8.7.3, "Cable Tray Sharing", have been reviewed and are not impacted by this modification. There are no cables being routed as part of this design, and fuses are being installed to protect against a cable fault in the cable tray. Section 8.7.7, "Panel Wiring Separation", is not impacted because wiring for redundant trains of ventilation are in separate terminal boxes.

Thus the evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98CO05 (Rev. 1) - Emergency Response Computer System (ERCS) Data Concentrators

Description of Change

The ERCS collects and processes selected field data for display to plant personnel. This information, in its multiple forms, is used to assist personnel in proper implementation of emergency operating procedures during an accident condition.

This modification replaced the existing data concentrator hardware and provided new software to replicate existing functionality. The software and equipment replicated the functionality of the existing pair of data concentrators on their new hardware, including

communication with the existing system. All the hardware and software now processes, sorts and recognizes date data between the twentieth and twenty-first centuries, which includes any leap year, without error in all date data output, reports, results or other values which rely on date definitions.

Summary of Safety Evaluation

The ERCS hardware is described within the USAR. Previously, the USAR explained that the ERCS redundant central processing units (CPU's) have data concentrators. The advancements of the new hardware allowed the existing two data concentrators to be combined into one. The previous software treated these existing data concentrators the same as the new single data concentrator. Therefore, the change of the description within USAR of "data concentrators" to "data concentrator" is a clarification of the description for ERCS and is not a functional change.

Thus the evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98EB02 – Repower Cooling Water System Common Unit Motor Valves

Description of Change

This modification repowered a number of cooling water motor-operated valves (MOV's) from transferable MCC's. Prior to the modification, these cooling water MOV's were common to both units, but were powered from non-transferable MCC's, such that a loss of power on one unit would cause the MOV's to be unpowered. The administrative controls associated with such an operational condition could be a burden to operators, particularly if power were removed from the MOV's for an extended period (e.g., outage maintenance of a power source.)

Summary of Safety Evaluation

All the repowered MOV's still have safeguards power supplies, with cables in analyzed environments. Also, there is no change in logic or function of the MOV's and the changes in electrical loading are within equipment ratings and load limits. Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98FP01 (Parts 1, 2, and 3) – MOV Hot Short

Description of Change

These modifications bring the MOV's into compliance with the plant Safe Shutdown Analysis and the current interpretation of Information Notice 92-18.

Part 1 rewired the "before open limit switch" of the Unit 1 RHR to reactor vessel (RV) Injection Isolation Valves A (MV-32064) and B (MV-32065) so that it is located in the open control circuit downstream of any potential hot short in one fire area that could spuriously open the valve. Moving the position limit switch contact (33/bo) to a location after the 42 open coil makes it an effective contact to stop the MOV motor for one or more hot shorts in one fire area anywhere in the open circuit.

Part 2 rewired the "after open limit switch" and the "closing torque switch" of the Unit 1 Reactor Coolant System (RCS) Loop A Hot Leg RHR Supply Valve (MV-32165) and the Unit 2 RCS Loop A Hot Leg RHR Supply Valve (MV-32193) so that they are located in the close control circuit downstream of any circuit failures in the Control/Relay Room fire area that could spuriously close the valve.

Part 3 relocated (schematically) limit switches in MOV control circuits so that they are located downstream of any circuit failures in the Control/Relay Room fire area that could spuriously operate each valve out of the safe shutdown credited position and damage the valve. The following valves were modified by this design change part:

MV-32238	11 AFW To 11 SG Motor-Operated Valve (MV)
MV-32246	22 AFW To 21 SG MV
MV-32333	11 TDAFWP Suction From Condensate Storage Tank (CST) MV
MV-32345	22 TDAFWP Suction From CST MV
MV-32084	11 Refueling Water Storage Tank (RWST) To 11 RHR Pump Isolation MV
MV-32187	21 RWST To 21 RHR Pump Isolation MV

Summary of 50.59 Evaluation

For all three parts, in accordance with the direction given in NRC Generic Letter 86-10, Part F, the seven questions (of 10CFR 50.59) address the postulated fire in the Fire Hazards Analysis and the Safe Shutdown Analysis for each fire area as an accident previously evaluated in addition to those in the USAR. The response to each of the seven questions for all three parts was "No." These modifications did not affect Technical Specifications or the USAR.

Modification 98RC03 – Repair of 22 Steam Generator Hot Leg Primary Manway Stud Holes 11 and 15 Using Threaded Inserts

Description of Change

A bolthole inspection was conducted on the Unit 2 steam generator primary manways and pressurizer manway as part of the Prairie Island corrective action process. Results of the inspection required repairs to holes number 11 and 15 on 22 steam generator hot leg primary manway under Design Change 98RC02. Both bolthole repairs using heli-coil

inserts were unsuccessful because the heli-coil thread holes were oversize. The oversize heli-coil holes were repaired using the installation of 1 and 7/8-inch bolthole threaded inserts with locking screw pins installed by this modification.

Summary of Safety Evaluation

The use of threaded inserts has been determined by Westinghouse to be in accordance with Section III of the ASME code to which the steam generator was designed. The margin of safety relating to primary pressure boundary integrity is dependent, in part, on the integrity of the primary manway closures. The margin of safety is consistent with the original design criteria, and the changes restore the closure structural and leakage integrity. The threaded inserts provide proper thread engagement and transfer of load to the manway pad and, therefore, manway closure integrity. Failure of individual threaded inserts might produce a leak past the gasketed surface of the closure. Catastrophic failure is bounded by the large break LOCA. Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98RV03 (Rev. 1) - Part Length Control Rod Drive Mechanism Cut/Unthread & Plug

Description of Change

The Prairie Island Nuclear Plant reactor vessel closure heads are equipped for part-length (P/L) control rods (4 locations per Unit). However, P/L rods have never been utilized, nor are there any plans to utilize them in the future. The P/L drive shaft lead screws are fully withdrawn and locked in place inside the motor tubes with anti-rotation devices. The P/L control rod drive mechanism (CRDM) power supply and rod position indication (RPI) coil stacks are disconnected.

Leakage was identified on a P/L motor tube assembly. The purpose of this Design Change is:

- Permanent removal of the P/L CRDM Head Adapter, Motor Tube and associated equipment by cutting the lower canopy seal weld and unthreading the head adapter from the head penetration.
- Permanent removal of the P/L CRDM motor stator/air baffle and RPI/seismic plate
- Installation of a threaded/seal welded head adapter plug.
- Installation of a "dummy" air baffle can to replace the motor stator/air baffle.
- Installation of a seismic spacer plate to replace the RPI top plate/seismic plate.

Following removal, the P/L CRDM leakage was evaluated for root cause determination. [See Prairie Island LER 2-98-02 for further details.]

Summary of Safety Evaluation

There are no Part Length Rod Assemblies inserted in the core. This Design Change will reduce the number of CRDM housings and therefore reducing the probability of a LOCA. In addition, at the affected locations, there are NO RCC assemblies; therefore rod ejection is not credible. The Part Length Rod locations are not relied on to mitigate the consequences of any accidents discussed in the USAR. Because there are no P/L Rod assemblies inserted in the core, these locations do not affect the RCC accidents discussed in the USAR. The CRDM Housing Mechanical Failure Evaluation takes credit for mitigation provided by the Full and Part Length RPIs. Following implementation of this Design Change, the CRDM housing will be removed, therefore the failure mechanism is eliminated from the these locations and the mitigation required by the RPI stack is no longer required. The installation of the new head adapter plugs will be in conformance with the applicable codes and standards (with exception of NRC approved alternate inspection). Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98RV06 (Parts 3 and 4) - RV Spare Penetration Canopy Seal Clamp Assembly Removal and Canopy Seal Weld Preventative Buildup

Description of Change

Part 3 of this modification removed the installed Canopy Seal Clamp Assemblies (installed under a separate modification) and preventatively buildup the remaining spare lower canopy seal welds at location E5 & I9 Unit 1, [I5, I9 Unit 2]. A special design will be provided for Unit 1 Location I9, which already has a 0.140" Westinghouse buildup. Unit 2 Location I5 was leaking.

Part 4 provided for a preventative buildup on all remaining Unit 1 & 2 Full Length CRDM intermediate canopy seal welds. The preventative buildup thickness will be reduced, if possible, to minimize exposure, and cost, while maintaining the canopy seal leak free thru remaining license life with a postulated 20-year life extension.

This modification affected only the leakage barrier provided by the canopy seal weld. The pressure retaining (structural) integrity of the joint between the head penetration and CRDM Latch Housing is provided by the threaded joint that joins the two components.

Summary of 50.59 Evaluation

Installation of the weld buildup would reduce the probability of canopy seal leakage (RCS leakage) by building up the affected seal weld. The weld buildup would not affect the structural integrity of the joint provided by the threads. CRDM housing mechanical

failures are evaluated in USAR Section 3.5.4.1. This Design Change would be in accordance with the applicable codes and standards (with exception of NRC approved alternate inspection). Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 98ZN02 – Isolate Control Room to Relay Room Vent

Description of Change

This design change isolated the control room ventilation system from the relay room ventilation by installing two blanks in the supply and return ducts for the relay room.

Summary of Safety Evaluation

The isolation of the relay room does not alter any assumptions used in the evaluation of the offsite radiological consequences of any accident discussed in the USAR. For control room doses, actual doses may be reduced due to the isolation from the relay room. Calculated control room doses are affected since the current design basis calculation evaluates both the relay room included and excluded conditions. The relay room not blanked off assumption results in higher control room dose. The environment assumed for equipment qualification is unchanged or less harsh. Other aspects of the environment (such as, oxygen levels, CO_2 levels, and toxic gas concentrations) could change within the relay room. However, these parameters do not affect the performance of equipment within the relay room, nor do they impair the CARDOX fire protection capability of the room. The Technical Specifications specify flow rates through the control room clean-up filters, maximum pressure drop across the filters and filter efficiencies. The isolation of the relay room will not affect the flow rate through the clean-up filters, the pressure drop or the filter efficiency. Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 99CG01 – Flow Meter/Totalizer in Nitrogen Line

Description of Change

This Design Change installed a flow meter/totalizer in the nitrogen supply line to each unit's pressurizer relief tank. This meter is used during reactor coolant system drain-down to allow operators to stop nitrogen injection after a predetermined quantity has been injected into the steam generator tubes. To ensure no meter bypass flow, a manual isolation valve will be installed in the parallel low flow nitrogen regulator line.

Summary of 50.59 Evaluation

The installed component is only intended for use during nitrogen injection to the steam generators, and does not perform any safety-related function. During all other plant conditions (power operation, shutdown modes, etc.) the component will typically be isolated.

The manual valve, installed in the low flow regulator line, does not impact safety. The low flow regulator is not relied on to mitigate the consequences of any accidents or malfunctions, and is not safety-related.

All of the questions to determine the need for NRC pre-approval were answered with the result that NRC pre-approval was not required.

Modification 99EB01 - Motor Control Centers 1T1/1T2 Transfer Switches

Description of Change

Motor Control Centers (MCC) 1T1 and 1T2 are transferable between two 480V sources, one from each unit. The T MCCs supply power to loads that are shared between Unit 1 and Unit 2. The 480V source breakers for the two sources for each MCC are the only isolation devices for the two cable feeds and are located at the 480 V bus end of the cable only. There are no breakers at the MCC ends of the cables; the cables are hard wired to the MCC main bus. Therefore, when the MCC is energized, both cable feeds are energized back to both 480V bus source breakers regardless of which bus is supplying the MCC.

There have been two near-miss events involving energized T MCC cable in an otherwise de-energized 480V bus. This design change will install a break-before-make transfer switch for each MCC. This switch will provide the necessary isolation to prevent the cable feed from the 480V bus that is not supplying the MCC from being energized via the MCC.

Summary of Safety Evaluation

The transfer switches are used as part of transferring their associated MCC between sources. The safety related (safeguards) function of the switch during energized operation of the MCC is to carry current. The switch is sized for the appropriate system voltage, ampacity, and short circuit withstand. The fuses installed as part of the switch (for short circuit capability) coordinate with the upstream and downstream protective devices so there is no effect on circuit coordination for the system. The switches and fuses are qualified to the same quality, seismic, and environmental qualifications as the electrical distribution system they are part of. There is no change in the safeguards operation, function, or qualification of the MCCs or their power sources. There are no new results of failure with the addition of the transfer switches. Therefore, the MCCs'

operation and their power sources' operation in previously evaluated accidents remains unchanged.

Transferring for maintenance or operational purposes will be done with the associated MCC de-energized as is done in the current design. The personnel safety function of the switch is to provide electrical isolation of the source cable that is not powering the MCC at the MCC end of the cable.

The work to implement this design change will be done in two stages:

1. Preparatory Work that can be performed without an MCC outage with both units at power, and

2. The cut-over Work that requires the MCC to isolated.

The final cutover wiring and testing of these MCC transfer switches is being done during MCC allowed out of service times to comply with Tech Spec requirements under a one time License Amendment that was processed for this design change.

This design change complies with generic letter (GL) 91-11 and Prairie Island's commitments thereto. This design change also retains the design features of the T MCCs as described in the USAR while adding measures to ensure better personnel protection during 480V bus maintenance. USAR Figure 8.3-1 will be revised to reflect the transfer switch additions as part of the design change process.

The evaluation concluded that the answer to the seven questions was "No," thus, prior NRC approval (beyond that addressed in the one-time commitment) was not required.

Modification 99FH02 - Repair Regions D, E and F Spent Fuel Assemblies

Description of Change

Prairie Island owns 238 Westinghouse Standard spent fuel assemblies that could not be handled using the normal spent fuel handling tool. These assemblies had degraded connections between the top nozzle and guide tubes. This Design Change repaired on the top nozzle connections to allow the fuel assemblies to be handled with the normal spent fuel handling tool.

Summary of Safety Evaluation

This modification repaired the top nozzle connection to meet or exceed the original mechanical design requirements. The original Final Safety Analysis Report states that the fuel assembly connection can be loaded axially to 2200 pounds with no damage resulting. Also, Exxon states that the dynamic loads for vertical handling shall be assumed to be equal to 2.5 times the dry assembly weight, which corresponds to 3150

pounds. The repaired connection after this modification can be loaded to 6 x 650 pounds or 3900 pounds. This exceeds both the original Westinghouse design and the Exxon design for the top nozzle connection.

The installation of anchors into the thimble tubes, which are similar to other inserts that have already been analyzed, will not cause any increase in reactivity in the spent fuel pool because the anchor displaces water, which reduces reactivity.

The following fuel handling accidents are evaluated in section 14.5.1 of the USAR:

- A fuel assembly becomes stuck inside the reactor vessel.
- A fuel assembly or control rod cluster is dropped onto the floor of the reactor cavity or spent fuel pool.
- A fuel assembly becomes stuck in penetration valve.
- A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

Fuel assemblies shall only be handled one at a time as presently assumed in the USAR. Also, repaired assemblies will not be reinstalled in the cavity and thus there will not be any reason to transport the assemblies to the cavity via the transfer system.

The accident that concerns this modification is a dropped fuel assembly in the spent fuel pool. This modification will repair the top nozzle in such a manner that the assembly can be moved utilizing the existing spent fuel handling tool. There will be no impact on the dropped assembly accident as analyzed in the USAR.

Considering all potential dynamic loads and fuel handling accidents, the strength of the modified nozzle connections exceeds the design requirements.

Based on the above discussion, the modification had no adverse impact on the safe operation of the plant and the associated licensing basis, nor does it present any new accident scenarios that would need to be analyzed.

This design change did not introduce any new failure modes for fuel handling in the Spent Fuel Pool not already discussed and analyzed in the Updated Safety Analysis Report.

Modification 99SG04 – Removal of Tube/Sleeve Samples from Unit 1 Steam Generator

Description of Change

This Design Change supported removal of ABB sleeves from steam generator 12 and/or steam generator tubes from Unit 1 as needed in order to meet the voltage based repair criteria on Unit 1 under Design Change 96SG05. The sleeve removal was to be used to gain additional knowledge about the degradation mechanisms of the PINGP steam generator tubing/sleeves. The sleeve removal was to be used to evaluate the effects on steam generator tube structural and leakage integrity of non-quantifiable eddy current indications. Removal of a sleeve was required during the April 1999 refueling outage. Additional tube samples were also to be possibly removed to meet the voltage based repair criteria or other unknown degradation mechanisms.

Welded tubesheet plugs were installed in the tubesheet bore holes left by the removed tubes and mechanical plugs in the opposite tube end.

Summary of Safety Evaluation

Since the plugs maintain pressure boundary integrity and leak tightness of the original tube, and since the failure modes are the same as for the tubing, there is no effect on the consequences of the MSLB, LOCA, or Steam Generator Tube Rupture (SGTR) accidents. The tube remnants left in the steam generator will not interact with adjacent tubes. The increase in borehole size does not unacceptably increase the tubesheet stress or reduce the fatigue margins. The tube removal and welded plug will not increase the radiological consequences of any of these accidents nor will it significantly degrade the ability of the steam generators to remove decay heat from the reactor coolant system. Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 99ZC01 (Rev. 1) – Airlock Door Shaft Seal Modification

Description of Change

In order to reduce the leakage from the airlock through the shaft seal assemblies the airlock door hand-wheel shaft seal assemblies were replaced and the rigid couplings were removed in order to:

- Increase the tolerance of the seal arrangement to handle moderate misalignment.

- Reduce shaft misalignment

The new housings can tolerate misalignment better than the previous for two reasons:

1. The O-rings are compressible; the Bal seals are not.
2. In order for the gasket between the existing seal housing and bulkhead to form an effective seal the housing must be bolted firmly against the airlock bulkhead.

This arrangement does not allow for any angular misalignment between the bulkhead and mechanism shaft. The assembly utilizes a flat gasket between the housing and airlock bulkhead. During installation the misalignment can be accounted for by increasing the torque on one or more housing mounting bolts thereby aligning the two. The compression rate around the perimeter of the gasket varies while maintaining an effective seal.

Summary of 50.59 Evaluation

Based on the evaluation of the seven questions of 10CFR 50.59, this design change did not present an unreviewed safety question. This modification did not affect the USAR or Technical Specifications.

Modification 00CL02 – Backup Air System for Cooling Water Control Valve

Description of Change

This modification installed a backup air system for the cooling water strainer backwash valves. There are two cooling water strainers per train of cooling water and their purpose is to remove particulate from the cooling water before it enters the cooling water header. The backwash cycle diverts up to 300 gallons per minute per strainer from the cooling water header. The instrument air header provides the motive force for the backwash valves. Air pressure maintains these valves closed. The valves fail open on a loss instrument air. This modification prevents the unnecessary diversion of flow from the cooling water header in the event of a loss of instrument air.

Summary of Safety Evaluation

The Design Change did not affect consequences (in terms of dose on-site or off-site). The cooling water system does not affect the source term used in dose analyses. The cooling water system does support equipment used to mitigate accidents, but the provision of the backup to instrument air to the cooling water strainers enhances system capability by preventing diversion of cooling water on a loss of instrument air.

The cooling water strainer functions are not accident precursors, so the probability of an accident is not changed by this design change. The design change increases the reliability of the backwash valves remaining closed when not required for backwashing and, in this respect, reduces the potentiai for malfunction.

The backup compressed air to the backwash valves preserves margins associated with the cooling water strainers by providing a redundant means of operating the backwash control valves. Therefore, the Design Change did not reduce the margin of safety as defined in the basis for any Technical Specifications.

Modification 00FH02 – Unit 1 Cycle 21 Core Reload

Description of Change

This modification replaced depleted Unit 1 fuel assemblies with a fresh reload of Westinghouse Vantage+ fuel assemblies allowing another cycle of power operation. All applicable documents and analyses were reviewed and performed for Unit 1 Cycle 21 assuring safe operation. The core design was verified by the performance of post-refueling startup physics testing.

Summary of Safety Evaluation

1. May the proposed activity increase the consequences of an accident previously evaluated in the USAR or in a pending USAR submittal?

   No. The consequences of all analyzed accidents have been reviewed and it has been determined that the Unit 1 Cycle 21 core will not increase the consequences of any accident previously analyzed. The analysis for offsite dose is valid for core average exposures less than 50,000 Megawatt Day/Metric Ton Unit (MWD/MTU) and maximum assembly exposures less than 75,000 MWD/MTU. The Unit 1 Cycle 21 core is projected to have a core average exposure of less than 42,000 MWD/MTU and a maximum assembly exposure of less than 62,000 MWD/MTU with the exception of T81, which is expected to reach an exposure of 70,030 MWD/MTU. The NRC has approved the burnup of T81 to beyond the 62,000 MWD/MTU limit. The radioactive inventory in the core is less than that used in the analysis and therefore would not increase consequences of an accident previously evaluated in the USAR or in a pending USAR submittal.

2. May the proposed activity increase the probability of occurrence of an accident previously evaluated in the USAR or in a pending USAR submittal?

   No. The only change to the plant resulting from this modification is the replacement of depleted fuel with fresh fuel, plus rearrangement of the fuel that will be reused in the core. The reactor core is not an initiator of accidents analyzed in the USAR therefore refueling will not change the probability of any accidents. Therefore, the modification to the core does not increase the probability of occurrence of an accident previously evaluated in the USAR or in a pending USAR submittal.

3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR or in a pending USAR submittal?

   No. The only change to the plant resulting from this modification is the replacement of depleted fuel with fresh fuel, plus rearrangement of the fuel that will be reused in the core. The new core satisfies all the design requirements stated in the USAR. Therefore, the core refueling will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR or in a pending USAR submittal.

4. May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR or in a pending USAR submittal?

   No. The consequences of all analyzed accidents have been reviewed and it has been determined that the Unit 1 Cycle 21 core will not increase the consequences of any accident previously analyzed. The analysis for offsite dose is valid for core average exposures less than 50,000 MWD/MTU and minimum assembly exposures less than 75,000 MWDMTU. The Unit 1 Cycle 21 core is projected to have a core average exposure of less than 42,000 MWD/MTU and a maximum assembly exposure of less than 70,030 MWD/MTU. The NRC has approved the burnup of T81 to beyond the 62,000 MWD/MTU limit. The radioactive inventory in the core is less than that used in the analysis and therefore would not increase consequences of any equipment malfunction previously evaluated in the USAR or in a pending USAR submittal.

5. May the proposed activity create the possibility of an accident of a different type than previously evaluated in the USAR or in a pending USAR submittal?

   No. The only change to the plant resulting from this modification is the replacement of depleted fuel with fresh fuel, plus rearrangement of the fuel that will stay in the core. The reactor core is not an initiator of accidents analyzed in the USAR therefore refueling will not create the possibility of a different type of accident. Therefore, the modification to the core does not create the possibility of occurrence of an accident of a different type than previously evaluated in the USAR or in a pending USAR submittal.

6. May the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR or in a pending USAR submittal?

   No. The only change to the plant resulting from this modification is the replacement of depleted fuel with fresh fuel, plus rearrangement of the fuel that will be reused in the core. The new core satisfies all the design requirements stated in the USAR. Therefore, the core refueling will not create the possibility of a different

type of malfunction of equipment important to safety than any previously evaluated in the USAR or in a pending USAR submittal.

7. Does the proposed activity reduce margin of safety as defined in the basis for any Technical Specification?

   No. The analyses for the new core show that the design criteria of departure from nucleate boiling ratio (DNBR), thermal and hydraulic capability, fuel temperature, cladding temperature, and cladding strain are all met. Therefore, the core refueling does not reduce the margin of safety as defined in the basis for any Technical Specification.

Modification 00FH03 - In-Mast Sipping on Manipulator Crane

Description of Change

In an effort to decrease critical path time and in essence reduce outage duration, failed fuel assembly identification is very critical. The current system requires additional fuel moves and time to perform the fuel sipping activities. This extends critical path fuel handling activities by 1-2 days.

The technology is available to allow the sipping procedure to be completed during normal fuel handling activities in the reactor core. This process will limit the number of assemblies that require canister system sipping.

This design change will modify the Manipulator Crane inner mast and tower assembly to accommodate the installation of the Westinghouse Windsor In-Mast Sipping System when required. The system will draw a water sample directly from the top of a fuel assembly, and then analyze the sample for Xenon and Krypton gases that would be present if a fuel rod contained a leak.

Summary of 50.59 Evaluation

This design change does not present an Unreviewed Safety Question Determination. Modification of the manipulator crane mast did not modify crane movement, the seismic design of the crane, nor will it increase the potential for damaging a fuel assembly during refueling activities. The design change will not have an adverse impact in the safe operation of the plant and the associated licensing basis, nor will it present any new accident scenarios that would need to be analyzed.

Modification 00FP01 (Part 1, Rev. 1) – Kaowool Removal

Description of Change

During the 1998 Fire Protection Functional Inspection (FPFI), the NRC concluded that the fire resistive performance of the Kaowool Fire Barrier System, used to comply with requirements of 10 CFR 50, Appendix R, is considered "indeterminate." A commitment to modify or replace the Kaowool Fire Barrier Systems was made in the October 8, 1998 response letter to the NRC.

The Kaowool Fire Barrier Systems used to meet the requirements of 10CFR50 Appendix R will be replaced with a different system. All the Kaowool Fire Barrier Systems, which are no longer required for Appendix R compliance, will be removed. The replacement system(s) will also be used to protect previously unwrapped raceways identified as requiring protection by the Appendix R Exemptions, the Safe Shutdown Analysis Revision and the response to Information Notice 92-18 (Hot Shorts).

The replacement system will be selected based on its ability to pass fire endurance tests conducted in accordance with Generic Letter 86-10, Supplement 1 to bound the configurations in the plant.

Revision 1 to this modification is required due to a change in scope to remove wrap requirements for:

- Unit 1 power-operated relief valve (PORV) inside containment (CV-31231 and CV-31232). Field measurements made during Unit 1 outage indicated that proper separation between PORV and Block Valve existed and wrapping was not required.

- Unit 2 PORV inside containment (CV-31233 and CV-31234). Field measurements made during Unit 2 outage indicated that proper separation between PORV and Block Valve did not exist. Cables for the PORV and Block Valves were re-routed under Modification 00FP01 – Part 2, therefore wrapping was not required under this part of the modification.

- Unit 1 and Unit 2 Vessel Injection Valves (MV-32065 and MV-32168). These valves were to be protected to prevent a flow diversion during RHR cooldown. Analysis has verified that these valves are not required to remain shut during cooldown and that we can still achieve cold shutdown conditions even if these valves would spuriously open, therefore the need to protect these valves is moot.

Revision 1 to this modification is required due to a change in scope to add additional wrap. The additional wrap is required due to the withdrawing of the exemption request for use of Rockbestos Appendix R cable in lieu of a 1 hour fire barrier that was installed under Modification 94L483 – Part G for the following components:

- 12 and 22 Charging Pump control cables located in FA 58 and 73.
- Unit 1 and Unit 2 PORV control cables located in FA 59 and 74.

Summary of Safety Evaluation

The safety evaluation resulted in no Unreviewed Safety Question. The USAR and the Technical Specifications do not require revision for the design change. The installation of the fire barrier system(s) and the removal of the Kaowool Fire Barrier System do not require taking any equipment out of service, nor will it affect the operation of any equipment. The design change provides compliance with the requirements of 10CFR50 Appendix R.

Modification 00FP01 (Part 2) – Kaowool Removal

Description of Change

This design change is in response to a condition report, which addresses the fact that the PORV and the Block valves may not be in compliance with Prairie Island's Exemptions from Appendix R Section III.G.2 for both Unit 1 and Unit 2 Containment. Therefore, measurements of the distance between the cables of concern for the PORV and the cables for the Block valves will be taken.

If the distance between the tray routed PORV cables and the block valve cables is less than the required 20', one or more of the cables and associated conduits will be relocated in order to be in compliance with the exemption.

Cables that are to be relocated will be determinated and pulled back far enough so the conduit can be rerouted to get the required separation. This can be done by routing PORV cabling in conduit where it is within 20' of any cables for the redundant Block valve. An alternative is to reroute the power and control cables for the Block valve so that they are at least 20' away from the tray routed PORV cables. The reason for this is the block valve has to operate from open to closed to be credited and the PORV is already closed and would need an external hot short to open. Prairie Island's Appendix R analysis assumes that a conduit is grounded and thus cannot pass an external hot short to an internal cable. If the PORV cables are in conduit within an Appendix R fire, there would be no way to get a hot short. Once this design change is done, PI will be back in compliance with the Containment exemption to Appendix R.

Summary of Safety Evaluation

The safety evaluation shows that this design change does not represent an Unresolved Safety Question (USQ) and as long as the design change is completed with the reactor in cold shutdown and with the Pressurizer manways removed, there are no LCO's which need to be entered.

Modification 00RC02 – Reroute RCS Hot Leg Drain Siphon Break

Description of Change

During completion of reduced inventory operations on May 23, 2000, the self-limiting drain path siphoned. Operators manually stopped the draining in accordance with procedure. A new temporary fan/filter assembly connected to the same ductwork as the siphon break caused the siphoning. The new assembly suction pressure was sufficient to prevent the siphon break from limiting the RCS draindown as designed.

This Design Change was implemented to reroute the common reactor coolant drain tank (RCDT) vent and siphon break line to containment atmosphere.

Summary of Safety Evaluation

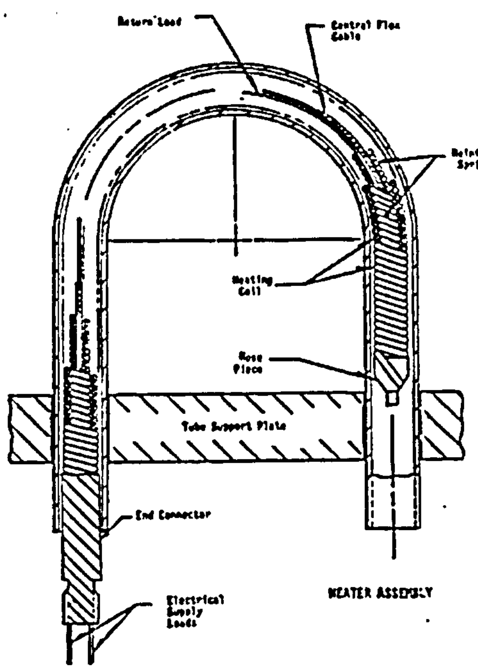
The siphon break is not credited in any of the accident analyses. During power operation the only connection to plant equipment is at the isolated connection to the RCDT. The vent/siphon break design functions during outages are not credited in any of the USAR analyses. Therefore, the probabilities or consequences of the USAR accidents are not increased. Likewise, there are no credible accidents that can be caused by the siphon break.

The siphon break does not affect any SSC that is considered important to safety. Therefore, the probability and consequence of a malfunction are not increased. Likewise, there are no credible malfunctions that can be caused by the siphon break that have not been previously considered.

Modification 00SG03 (Rev. 1) - Heat Treat SG Rows 1&2 U-Bends

Description of Change

Primary water stress corrosion cracking (PWSCC) of mill annealed Alloy 600 steam generator tubing has been identified as having a potentially significant impact on steam generator availability. This concern was highlighted by the large leak event at Indian Point 2 on February 15, 2000. The Prairie Island steam generators utilize the same type mill annealed Alloy 600 non-stress relieved tubing as Indian Point 2.



Once a steam generator is placed in service, the most effective means for minimizing the propensity for PWSCC is to reduce or modify the residual stress in the regions of the tube most prone to attack. One method known to reduce residual stress through the tube wall, from inside diameter to outside diameter, is by heating the tubing to temperatures above 1100°F for a sufficient time.

This Design Change does NOT affect Technical Specifications.

The proposed heat treatment does not adversely affect the strength or integrity of the SG tubes. Based on a review of the Design Inputs, and the PINGP USAR, there are no safety concerns created by the proposed heat treatment.

Summary of Safety Evaluation

The Safety Evaluation section of this Design Change reviewed the affect of heat treatment on the plant design basis accidents and determined there is no unreviewed safety question.

Modification 00SG04 - Removal of Tube/Sleeve Samples from Unit 1 Steam Generator

Description of Change

Design Change 97SG05 (License Amendments 133 and 125) implemented voltage based repair criteria (ARC) in accordance with GL 95-05. Initial tube pulls required for implementation of the ARC were completed in October 1997 under Design Change 96SG04.

As reflected in the LAR and NRC Safety Evaluation, NSP committed to future tube pulls consistent with the tube removal guidelines in GL 95-05 until an NRC-endorsed industry program is available. The NRC endorsed Electric Power Research Institute (EPRI) Steam Generator Degradation Specific Management (SGDSM) Database and accepted follow-up tube pulls beginning three operating cycles following the previous tube pull. Therefore, follow-up tube pulls need not be considered during the January 2001 outage.

However, the EPRI program added the following criteria, also accepted by the NRC.

> *If indications with unanticipated voltage substantially higher than the structural limit (for example, >10 volts) from the burst correlation are found in an inspection, the indication should be considered for removal and destructive examination if the test results are likely to determine whether or not condition monitoring or operational assessment results would satisfy acceptance limits.*

This design change is to support the removal of ABB sleeves from steam generator 12 and/or steam generator tubes from Unit 1 as may be necessary for the EPRI program or to characterize unusual indications of degradation. This design change provides for installation of an oversized bore hole, and installation of ABB Combustion Engineering (ABB / CE) welded tube sheet plugs and mechanical plugs at the pulled tube locations.

Summary of Safety Evaluation

The safety evaluation addresses the removal of tube and sleeve samples, installation of an oversized bore hole, and installation of ABB / CE welded tubesheet plugs and mechanical plugs. There is no unreviewed safety question.

Modification 00SI01 (Part 1, Rev. 1) – Boric Acid Reduction

Description of Change

This safety evaluation addresses the aspects of the modification not covered by the Safety Evaluation issued by the NRC for License Amendments 156 and 147 for Unit 1 and Unit 2, respectively. Primarily, this evaluation addresses the preoperational testing planned after the installation of Parts 1A and 1B of the modification.

Part 1A installation occurred at-power shortly after the NRC approved the associated License Amendment. Most activities for this part involved positioning of valves in the desired configuration and removing power to the MOV's by opening breakers. Preoperational testing for this part involved stroking MV-32079 and MV-32080 to verify operability of the "SI Not Ready" lights on the Main Control Board. Power was removed from the valves after testing was complete.

Part 1B installation occurred during a refueling outage. Activities included control circuit changes for the MOVs, status light changes, removal of two boric acid storage tank

(BAST) level instrument loops and the logic associated with swap-over from the BAST to the RWST, and removal of second starters from MV-32079 and MV-32080. Once again, preoperational testing involved stroking MV-32079 and MV-32080 to verify operability of the "SI Not Ready" lights on the Main Control Board. Power was removed from the valves after testing was complete.

Summary of Safety Evaluation

Thus, the safety evaluation concluded that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 00VC01 – Removal of Heat Trace Alarms

Description of Change

Prior to the modification, all heat trace circuits annunciated in the control room for the following conditions:

1. Low temperature.
2. High temperature.
3. Heater failure.

All circuits are also alarmed on local panels in the Auxiliary Building. Operations personnel monitor these panels several times during each shift.

During make-up's to the volume control tank using the blender, the reactor make up water used is sufficiently below the low temperature alarm setpoint of 160°F, such that, the control room received a low temperature alarm on two heat trace circuits that remained in until the line reheated. The low temperature in the line had no effect as far as crystallization is concerned, as the blended flow is a very low concentration of boric acid mixed with the reactor make-up water. Prairie Island Operations classified these low temperature alarms on heat trace as nuisance alarms due to the high frequency of this evolution being performed. The low temperature alarms still alarm locally in the Auxiliary building allowing the Auxiliary Building operators to be aware of and monitor them.

This Design Change lifted four leads in the local low temperature alarm panel that feed the control room alarm. The circuitry was modified so no other alarms were affected.

Summary of Safety Evaluation

Based on the evaluation of the seven questions of 10 CFR 50.59, this design change did not present an unreviewed safety question. This modification did not affect Technical Specifications, but did result in a change to the USAR.

Modification 01FH02 - Unit 2 Cycle 21 Core Reload

Description of Change

This design change is required to allow for continued power operation of Prairie Island Unit 2 for approximately 19 months. The fuel in the current core has been burned to a state that no longer allows for full power operation. This reload will replace burned fuel from Unit 2 Cycle 20 with fresh fuel from Westinghouse as well as some fuel used in previous cycles at Prairie Island. This will allow Unit 2 to produce power at its rated capacity.

Summary of 50.59 Evaluation

The 50.59 Evaluation Concluded:

The core has been designed according to NRC approved methodology and transient analysis has been performed for all accidents in Chapter 14 of the USAR that need to be addressed. These transient analyses were performed using NRC approved methodology as well.

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 21 is safe and consistent with Prairie Island's current Licensing Basis.

Basis of Determination:

> Does the activity result in more than a minimal increase in the frequency of occurrence of an accident?
>
> No. The only change to the plant resulting from this modification is the replacement of burned fuel with fresh fuel and other burned fuel, plus rearrangement of the fuel that will be reused in the core. The reactor core is not an initiator of accident analyzed in the USAR so refueling will not change the probability of any accidents occurring. Therefore the modification to the core does not result in more than a minimal increase in the frequency of occurrence of an accident.

Does the activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety?

No. The only change to the plant resulting from this modification is the replacement of burned fuel with fresh fuel and other burned fuel, plus rearrangement of the fuel that will be reused in the core. The new core satisfies all the design requirements stated in the USAR. Therefore the core refueling will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

Does the activity result in more than a minimal increase in the consequences of an accident?

No. The consequences of all analyzed accidents have been reviewed and it has been determined that the Unit 2 Cycle 21 core will not increase the consequences of an accident. The analysis for offsite dose is valid for core average exposures less than 50,000 MWd/MTU and maximum assembly exposures less than 75,000 MWd/MTU. The Unit 2 Cycle 21 core is projected to have a core average exposure of less than 41,000 MWd/MTU and a maximum assembly exposure of less than 57,000 MWd/MTU. The radioactive inventory of the core is less than that used in the offsite dose analysis and would not increase the consequences of an accident. Therefore the activity will not result in more than a minimal increase in the consequences of an accident.

Does the activity result in more than a minimal increase in the consequences of a malfunction?

No. The consequences of all malfunctions have been reviewed and it has been determined that the Unit 2 Cycle 21 core will not increase the consequences of an accident. The analysis for offsite dose is valid for core average exposures less than 50,000 MWd/MTU and maximum assembly exposures less than 75,000 MWd/MTU. The Unit 2 Cycle 21 core is projected to have a core average exposure of less than 41,000 MWd/MTU and a maximum assembly exposure of less than 57,000 MWd/MTU. The radioactive inventory of the core is less than that used in the offsite dose analysis and would not increase the consequences of a malfunction. Therefore the activity will not result in more than a minimal increase in the consequences of a malfunction.

Does the activity create a possibility for an accident of a different type?

No. The only change to the plant resulting from this modification is the replacement of burned fuel with fresh fuel and other burned fuel, plus rearrangement of the fuel that will be reused in the core. The reactor core is not an initiator of accident analyzed in the USAR so refueling will not create the possibility of a different type of accident. Therefore the modification to the core does not create a possibility for an accident of a different type.

Does the activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated)?

No. The only change to the plant resulting from this modification is the replacement of burned fuel with fresh fuel and other burned fuel, plus rearrangement of the fuel that will be reused in the core. The new core satisfies all the design requirements stated in the USAR. Therefore the core refueling will not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated).

Does the activity result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered?

No. The analyses for the new core show that the design basis limits for DNBR, fuel temperature, fuel enthalpy, failed fuel pins, cladding temperature, cladding strain, RCS pressure, main steam pressure, steam generator differential pressure, and containment pressure are all met. Clad oxidation, RCS boundary stresses, and RCS boundary heat-up/cool-down are not subject to evaluation for this design change. Therefore the core refueling does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered.

Does the activity result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analysis?

No. The new core design was evaluated using the same methods as referred to in Section 14 of the USAR. The core design methods that support the safety methods have been updated and were approved by the NRC. Therefore the new core reload does not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analysis.

Modification 01RH01 - Residual Heat Removal Discharge Press Loop 1E/Non-1E Separation

Description of Change

This design change and evaluation was to resolve the apparent discrepancy between the RHR discharge pressure loop instruments for both trains on both Unit 1 and Unit 2, which were not designed to Institute of Electrical and Electronics Engineers (IEEE) 279 standards as a safety related function of the RHR system as should be, and the interlock function, which is for a safety related valve used to mitigate the consequences of certain Small Break Loss of Coolant Accidents (SBLOCA). As part of the evaluation, it was shown that the current loop design is consistent with the original plant design and licensing basis and that the plant (like other Westinghouse pressurized water reactors of

this vintage) relies on local operator action to open the valves upon failure of the instrument loops. This design change and evaluation also reviewed the other administrative and procedural controls on these valves, which prevent inadvertent operation of the valves and justified removing the instrument loop pressure interlock from the valve control circuits. This prevents the loop failure from forcing local operator action and allows operation of the valve from the control room. This allowed downgrading the electrical functions of the loop to all non-safety related indication.

Summary of Safety Evaluation

The design function of the loop interlock is to prevent inadvertent opening of the associated isolation valve for RHR system pressure greater than 210 psig, which is the SI pump suction piping rating. This function is not described in the USAR, however, it is discussed in the PINGP and Westinghouse Design Bases Documents and can be inferred from the design information available (logic drawings, specifications, and USAR statements about piping rating).

The loops each provide a pressure interlock (open permissive) to open the associated isolation valves when RHR discharge pressure is less than 210 psig. The valves are used to align the RHR system as a source to the SI system for high head recirculation following a SBLOCA. As such, the loops can affect an accident mitigation function as described in USAR sections 6.2.2.1.2 and 10.2.4.

If the valves were opened in error when pressure was too high, this could result in overpressurizing the SI suction piping. This may lead to opening a relief valve on the SI piping in the auxiliary building thereby causing a release of whatever water is in the line. The 1-inch line may not have adequate capacity to relieve pressure capacity of an RHR pump. Such an intersystem LOCA may be considered an accident initiation although there is no such design basis accident analyzed in the USAR.

This evaluation documents that the original design basis for these loops was non-safety related and that manual operator action was relied on because they cannot be credited with performing their function because they do not meet IEEE 279. Given that design basis, the evaluation and design change justified removing the interlock from the valve control circuit to allow remote operation for all scenarios based on the existing administrative and procedural controls for these valves.

Thus the response to each of the seven questions for all three parts was "No" concluding that this modification did not require NRC review and approval and did not affect or change Plant Technical Specifications.

Modification 01ZX01 (Part 2, Rev. 1) – Replace ZX Piping with Coated Carbon Steel and Super Austenitic Stainless Steel

Description of Change

This modification replaced and upgraded the ZX piping and cooling coils along with system operational enhancements to prevent microbiologically induced corrosion (MIC) and restored heat removal capabilities. The ZX system is a common non-safety related chilled water system added for both Units 1 and 2. The system provides chilled water during normal plant operation in the summer for the containment air cooling fan coil units and the control rod drive mechanism shroud cooling coils, as well as unit coolers located in the auxiliary building. The system is integrated with the cooling water system. The combined system provides chilled water during normal plant operation, but on a loss of power or safety signal actuation, the system isolates the chilled water system from the cooling water system.

Summary of 50.59 Evaluation

The activity of installing, service use, and evaluated failure effects of a qualified internal coating on ZX system piping has been evaluated per 10CFR50.59. The evaluation determined the activity does not require prior NRC approval and is safe to perform.

Modification 02CL01 – Bearing Water to Cooling Water Pumps

Description of Change

This design change provides an additional source of bearing water to the safeguards cooling water pumps (12 and 22 DDCLPs and 121 MDCLP). Currently, there are independent safety related supplies of bearing water to each safeguards cooling water pump that serves both the upper and lower bearings. The supply is from the cooling water headers, downstream of each respective pump's discharge check valve. This water is essentially river water that contains silt and sand. The pump vendor, in a letter dated November 6, 2000, recommended only limited operation of the pump on unfiltered river water for a period not to exceed 1000 hours without a complete internal inspection. For this reason, two parallel filters, one in service and one in standby, have been provided for each pump to enhance the cleanliness of the water prior to it entering the bearings. However, these filters clog frequently during periods of heavy barge traffic on the Mississippi and present an operations concern. This modification will connect the non-safety related clean filtered/well water supply to each pump's bearing water system to serve as the normal supply of bearing water. This system will have a mechanical three-way valve that will automatically switch to the safety related supply when there is a reduction in pressure from the well/filtered water system below the determined setpoint.

In addition to installing the new clean water source to the safeguards cooling water pump bearings, this modification will implement various enhancements to the safety-related

bearing water supply systems and the well/filtered water system as described in this document.

Summary of 50.59 Evaluation

A 10CFR 50.59 Evaluation was performed and addresses the additional source of clean water to the safeguards cooling water pumps. The evaluation has determined that NRC approval is not required.

Modification 02D101 – Isolate D1/D2 Coolant System Crossflow

Description of Change

This design change:

- Resolved the engine coolant crossflow from the jacket coolant loop to the air coolant heat exchanger loop by the addition of isolation valves and air release vents.
- Facilitates sampling and venting of the engine coolant system by extending the sample/vent line and adding a new sample valve in a more accessible area of the engine skid.
- Eliminates the potential for chromated water from the D1 engine coolant system being released to the Turbine Building sump by removing a portion of the D1 expansion tank overflow pipe and rerouting to the floor of the D1 diesel generator room.
- Completion of this Design Change will allow for closure of Temporary Modification 01T092

Summary of 50.59 Evaluation

Since this design change activity provides for the addition of new materials and components, a 10CFR 50.59 Evaluation was required to be performed. The result of this evaluation was that an NRC approval, prior to design change implementation, was not required for this design change.

Modification 02FH02 - Unit 1 Cycle 22 Core Reload

Description of Change

This Design Change is required to allow for continued power operation of Prairie Island Unit 1 for approximately 21 months. The fuel in the current core will be burned to a state

that no longer allows for full power operation. This reload will replace burned fuel from Unit 1 Cycle 21 with fresh fuel from Westinghouse as well as some fuel used in previous cycles at Prairie Island. This will allow the Unit 1 reactor to produce power at its rated capacity.

This design change will also allow for the replacement of W50's top nozzle, which has the potential to have faulty spring screws. Westinghouse, who is the design organization, will provide and install the replacement top nozzle with new spring screws that has shown to be less susceptible to spring screw fracture.

Summary of 50.59 Evaluation

The core has been designed according to NRC approved methodology and transient analyses have been performed for all accidents in Chapter 14 of the USAR that need to be addressed. These transient analyses were performed using NRC approved methodology.

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 22 is safe and consistent with Prairie Island's current Licensing Basis.

The top nozzle replacement for assembly W50 is a Westinghouse endorsed change. Based on Westinghouse analyses, the replacement top nozzle meets all the design and functional requirements for the original top nozzle without any loss of load capability and does not adversely impact the form, fit, or function of the fuel assembly. In fact, the original top nozzle's design parameters bound the new top nozzle's design. In addition, using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses. Therefore, the top nozzle replacement is safe and consistent with Prairie Island's current Licensing Basis.

Basis of Determination:

> Does the activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR?
>
> No. The only changes to the plant resulting from this modification are the replacement of burned fuel with fresh fuel and other burned fuel, rearrangement of the fuel that will be reused in the core and the replacement of a fuel assembly top nozzle. Neither the reactor core nor top nozzle replacement are initiators of accidents analyzed in the USAR so they will not change the probability of any accidents occurring and they will not introduce any new failure modes. Refueling the core is within the design and regulatory bases as stated in the USAR. The assemblies that will be used in the core meet the same fuel assembly design standards as in previous cores and will be handled using the standard fuel handling equipment. In addition, the operating Unit 1 Cycle 22 core meets all the

acceptance criteria for the Chapter 14 accidents. Changes to the top nozzle were made to prevent loose parts from entering the RCS and to ease fitting up the new top nozzle to an irradiated assembly. Even with these changes, the new top nozzle meets the design requirements of the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. Since these activities satisfy the necessary design requirements, the assumptions used in any of the safety analyses are not impacted and the frequency of occurrence of an accident is not increased.

Does the activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

No. The only changes to the plant resulting from this modification are the replacement of burned fuel with fresh fuel and other burned fuel, rearrangement of the fuel that will be reused in the core and the replacement of a fuel assembly top nozzle. Core refueling satisfies all the design requirements stated in the USAR. Therefore the core modification will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. In addition, the operating Unit 1 Cycle 22 core meets all the acceptance criteria for the Chapter 14 accidents.

Replacing W50's top nozzle is a further means of protection that loose parts will not be introduced into the RCS and thus cause SSC malfunctions. Changes to the top nozzle were made to prevent loose parts from entering the RCS and to ease fitting up the new top nozzle to an irradiated assembly. Even with these changes, the new top nozzle meets the design requirements of the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. In addition to design requirements, the replacement top nozzle meets all the functional requirements for the original top nozzle and does not adversely impact the form, fit, or function of the fuel assembly. Also, using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses. Since top nozzle replacement satisfies the necessary design requirements, the assumptions used in any of the safety analyses are not impacted and the likelihood of an occurrence of a malfunction of an SSC important to safety is not minimally increased.

Does the activity result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR?

No. The consequences of all analyzed accidents have been reviewed and it has been determined that the Unit 1 Cycle 22 core will not increase the consequences of an accident. The analysis for offsite dose is valid for core average exposures less than 50,000 MWd/MTU and maximum assembly exposures less than 75,000 MWd/MTU. The Unit 1 Cycle 22 core is projected to have a core average

exposure of less than 41,000 MWd/MTU and a maximum assembly exposure of less than 59,000 MWd/MTU. The radioactive inventory of the core is less than that used in the offsite dose analysis and would not increase the consequences of an accident. Therefore core reloading will not result in more than a minimal increase in the consequences of an accident.

Replacing W50's top nozzle does not introduce the possibility of a change in the consequences of an accident because top nozzle replacement is not an initiator of accidents and no new failure modes are introduced. Changes to the top nozzle were made to prevent loose parts from entering the RCS and to ease fitting up the new top nozzle to an irradiated assembly. Even with these changes, the new top nozzle meets the design requirements of the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. In addition to design requirements, the replacement top nozzle meets all the functional requirements for the original top nozzle and does not adversely impact the form, fit, or function of the fuel assembly. Using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses because the change does not affect any normal plant operating parameters, safeguard system actuation, or the assumptions and input parameters used in those analyses. Since top nozzle replacement satisfies the necessary design requirements and the radioactive inventory of the fuel assembly is not changed, the assumptions used in any of the safety analyses are not impacted and the consequences of an accident are not increased.

Does the activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR?

No. The consequences of all malfunctions have been reviewed and it has been determined that the Unit 1 Cycle 22 core will not increase the consequences of an accident. The analysis for offsite dose is valid for core average exposures less than 50,000 MWd/MTU and maximum assembly exposures less than 75,000 MWd/MTU. The Unit 1 Cycle 22 core is projected to have a core average exposure of less than 41,000 MWd/MTU and a maximum assembly exposure of less than 59,000 MWd/MTU. The radioactive inventory of the core is less than that used in the offsite dose analysis and would not increase the consequences of a malfunction. Therefore the activity will not result in more than a minimal increase in the consequences of a malfunction.

Replacing W50's top nozzle does not introduce the possibility of a change in the consequences of a malfunction because top nozzle replacement is not an initiator of any new malfunctions and no new failure modes are introduced. Changes to the top nozzle were made to prevent loose parts from entering the RCS and to ease fitting up the new top nozzle to an irradiated assembly. Even with these changes, the new top nozzle meets the design requirements of the original top nozzle without any loss of load capability. In fact, the original top nozzle's design

criteria bound the new top nozzle's design. In addition to design requirements, the replacement top nozzle meets all the functional requirements for the original top nozzle and does not adversely impact the form, fit, or function of the fuel assembly. Using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses because the change does not affect any normal plant operating parameters, safeguard system actuation, or the assumptions and input parameters used in those analyses. Since top nozzle replacement satisfies the necessary design requirements and the radioactive inventory of the fuel assembly is not changed, the consequences of any malfunctions are already bounded by the current analyses. Therefore, top nozzle replacement does not result in more than a minimal increase in the consequences of a malfunction.

Does the activity create a possibility for an accident of a different type than previously evaluated in the UFSAR?

No. The only changes to the plant resulting from this modification are the replacement of burned fuel with fresh fuel and other burned fuel, rearrangement of the fuel that will be reused in the core and the replacement of a fuel assembly top nozzle. Neither the reactor core nor top nozzle replacement is initiators of accidents analyzed in the USAR, so they will not create the possibility of a different type of accident. Nor does either activity introduce any new failure modes. Since both modifications meet the necessary design requirements, any type of accident that could occur as a result of modifying the core or replacing a top nozzle is bounded by the fuel handling accident (USAR Section 14.5.1). In addition, the operating Unit 1 Cycle 22 core meets all the acceptance criteria for the Chapter 14 accidents. Therefore the core modification, core operation, and the top nozzle replacement do not create a possibility for an accident of a different type.

Does the activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR?

No. The only changes to the plant resulting from this modification are the replacement of burned fuel with fresh fuel and other burned fuel, rearrangement of the fuel that will be reused in the core and the replacement of a fuel assembly top nozzle. The new core satisfies all the design requirements stated in the USAR and does not introduce new failure modes. Changes to the top nozzle were made to prevent loose parts from entering the RCS and to ease fitting up the new top nozzle to an irradiated assembly. Even with these changes, the new top nozzle meets the design requirements of the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. In addition to design requirements, the replacement top nozzle meets all the functional requirements for the original top nozzle and does not adversely impact the form, fit, or function of the fuel assembly. Lastly, top nozzle replacement does not introduce any new failure modes. Therefore the core

refueling and top nozzle replacement will not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the updated safety analysis report.

Does the activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

No. The analyses for the new core show that the design basis limits for DNBR, fuel temperature, fuel enthalpy, failed fuel pins, cladding temperature, cladding strain, RCS pressure, main steam pressure, steam generator differential pressure, and containment pressure are all met. Clad oxidation, RCS boundary stresses, and RCS boundary heat-up/cool-down are not subject to evaluation for this design change. Therefore, the core refueling does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The new top nozzle has been designed to fit up to the existing fuel assembly and meets all the design and functional requirements for the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. In addition, the new top nozzle does not adversely impact the form, fit, or function of the fuel assembly. Using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses because the change does not affect any normal plant operating parameters, safeguard system actuation, or the assumptions and input parameters used in those analyses. Therefore, top nozzle replacement will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

Does the activity result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis?

No. The new core design was evaluated using the same methods as referred to in Section 14 of the USAR, thus this question is not applicable. The core design methods that support the safety methods have been updated and were approved by the NRC. Therefore the new core reload does not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analysis.

Top nozzle replacement modifies assembly W50 and does not constitute a change to a method of evaluation defined in the USAR, thus this question is not applicable. The new top nozzle has been designed to fit up to the existing fuel assembly and meets all the design and functional requirements for the original top nozzle without any loss of load capability. In fact, the original top nozzle's design criteria bound the new top nozzle's design. In addition, the new top nozzle does not adversely impact the form, fit, or function of the fuel assembly. Using a fuel assembly with a replacement top nozzle does not affect the performance of any safety related system or negatively impact any analyses because the change does not affect any

normal plant operating parameters, safeguard system actuation, or the assumptions and input parameters used in those analyses. Therefore, top nozzle replacement does not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282. 50-306

December 10, 2003

CHANGES TO REGULATORY COMMITMENTS

2 pages follow

CHANGES TO REGULATORY COMMITMENTS

Regulatory Commitment Change 03-01

In response to NRC Generic Letter 89-19, Northern States Power (NSP) committed to revise the feedwater isolation Surveillance Procedures to verify that control valves move from full open to full closed in five seconds or less upon de-energizing either air supply solenoid valve. This limit was put in place to protect against steam generator overfill. Analysis has shown that, conservatively, the valves can take over 50 seconds to close and still protect against steam generator overfill. The current main steam line break (MSLB) analysis of record assumes that the valves will close in seven seconds. The requirement to verify the isolation time of each Main Feedwater Regulating Valve (MFRV) and MFRV bypass valve is within limits of the MSLB analysis of record has been incorporated in the Prairie Island Technical Specifications as SR 3.7.3.1, thus, the commitment is no longer necessary.

Regulatory Commitment Change 03-02

By letters dated June 30, 1982 and October 22, 1982, NSP requested exemption from the requirements of 10CFR 50, Appendix R, Section III.G.2, in Fire Area 31. As part of the approved exemption, NSP committed to install a one-hour fire barrier (Kaowool) to protect Train B safe shutdown conduits and install a thermal shield on the top and bottom of cable tray 2SG-LB17.

A review of the approved exemptions against the current safe shutdown analysis was completed and (the results transmitted in LER 1-98-12, Supplement 3). The results of the review identified that various design changes in the mid-90's removed some of the Kaowool fire barriers and the thermal shield, and manual actions were credited in the safe shutdown analysis. To correct these discrepancies, it was recommended that the 1-hr barrier be re-installed around one train of systems. Because a majority of the Train A AFW systems components are located outside Fire Area 31, it was determined that protecting Train A safe shutdown cables (in lieu of Train B) with a 1-hr fire barrier met the intent of Section III.G.2.

A modification removed the remaining Kaowool fire barriers and replaced them with 3M fire barriers (1-hour fire rating), and also installed the 1-hr rated protection (3M fire barrier) around the Train A cables that would require protection from fire damage. This modification was completed in 2002.

This commitment change only addresses the commitment specifically made in Fire Area 31 to protect Train B safe shutdown conduits with a 1-hr fire barrier and to install a thermal shield on top and bottom of one cable tray 2SG-LB17. Other commitments made in the approved exemption are unchanged.

Regulatory Commitment Change 03-03

The NRC Safety Evaluation for Prairie Island License Amendment 137 (Unit 1) and 128 (Unit 2), section 3.1, subheading "Repair Methodology", third paragraph, second sentence, notes: To address this issue, the licensee implemented several changes In the F* repair procedure that are similarly applied to the EF* repair process that will minimize the potential for returning tubes to service by rerolling with unacceptably low resistance to leakage. As described in CEN-620-P, Revision 05-P, the licensee will perform bobbin coil profilometery to ensure a minimum expansion level was achieved in the rerolling process. The revised repair procedure will also add an additional roll expansion over the original roll transition zone. A third change includes the performance of a secondary side hydrostatic test to verify the leakage integrity of newly installed rerolls. These measures provide assurance that tubes returned to service via the EF* repair criterion will have adequate leakage integrity during normal operating and postulated accident conditions.

It appears that, in the NRC Safety Evaluation for Amendment 137/128, the NRC inferred a commitment to complete secondary side hydrostatic testing of newly installed rerolls. While NSP did not formally commit to this, the following change has been processed under the Prairie Island Commitment Change process: Because in the preceding five outages (in which there have been over 1800 rerolls) the secondary side pressure test has not resulted in the rejection of a single reroll, NMC will no longer perform secondary side hydrostatic pressure testing of newly installed rerolls.