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Section 50.71

U S Nuclear Regulator" Commission
Attn: Document Controi Desk
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Submittal of Revision No. 12 to the
Updated Safety Analysis Report (USAR)

Pursuant to 10 CFR 50.71(e) we are submitting one original and 13 copies of Revision No. 12 to the Updated Safety Analysis Repcrt (USAR) for the Prairie Island Nuclear Generating Plant. This revision updates the information in the USAR up through December 16, 1993 (although some information is more recent).

Exhibit A contains descriptions and summaries of the safety evaluation for changes, tests and experiments made under the provisions of 10 CFR 50.59 during the period since the last update.

Exhibit B contains the USAR page changes and instructions for entering the pages.

Included in Exhibit B is Revision 17 to the Northern "tates Power Company Operational Quality Assurance Plan in compliance with 10 CFR 50.54(a). Changes in Revision 17 to the Plan are described in Appendix D to the Operational Quality Assurance Plan (which i"self is Appendix C to the USAR).

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for
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c: Regional Administrator - Region III, NRC
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Attachments: Exhibit A
Exhibit B

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Exhibit A

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
ANNUAL REPORT OF CHANGES, TESTS AND EXPERIMENTS - DECEMBER, 1991

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

1. Safety Evaluation 368 - Removal of Refueling Water Storage Tank (RWST) to Containment Spray Pump (CSP) MOVs from the GL 89-10 Program (MOV Testing)

   Description of Change

   The CSP is permanently aligned to the RWST. With only one suction source necessary, the CSP suction from the RWST MOVs are disabled in the open position, and the valves are removed from the GL 89-10 program. Applicable MOVs are MV-32098, 32099, 32110,32111.

   Summary of Safety Evaluation

   Safety Evaluation 364 concluded that containment spray is not needed for accident mitigation during the recirculation mode. The Residual Heat Removal System to CSP suction MOVs were disabled in the closed position.

   Since only the RWST to CSP suction is necessary, the other possible safety functions of the RWST to CSP isolation MOVs are evaluated. The valves are not required to prevent radioactive contamination of the RWST, nor to block flow from RHR to the RWST. The conclusion is that the MOVs can be maintained open and do not need to close as part of any accident response. The requirements of GL 89-10 can now be implemented. The MOVs must be prevented from possible misposition from the control room and removed from the EOPs.

2. Safety Evaluation 364 - Removal of Residual Heat Removal (RHR) to Containment Spray Pump (CSP) MOVs from the GL 89-10 Program (MOV Testing)

   Description of Change

   Analysis has determined that CS using recirculation flow from the RHR system is not necessary. Any reference to CS using RHR recirculation is removed from the USAR. MOVs from the RHR system to the CSP suction are permanently closed and removed from the GL 89-10 program. The affected valves are MV-32096, 32097, 32108, and 32109.

   Summary of Safety Evaluation

   A previous Safety Evaluation (No. 234) concluded that CS is not required in the recirculation mode. Sump pH control, containment pressure control, iodine removal, and environmental qualification (EQ) USAR analyses are not affected by the CS recirculation mode deletion. Additionally, the LOCA analyses do not use CS recirculation mode as an input. Any reference to CS recirculation mode operation can now be removed from the USAR.

The Safety Evaluation supplies the justification for removing the RHR to CSP recirculation mode of operation from the EOPs. Therefore, the RHR to CSP suction MOVs do not need to function during an accident and can be removed from the GL 89-10 program after they are permanently removed from service.

3. Safety Evaluation 366 - Main Control Board Separation Criteria

Description of Change

There are five cases where as-built configuration on the Main Control Board does not meet the USAR design criteria. The Safety Evaluation justifies the acceptability of these cases and the expansion of the USAR design criteria.

Summary of Safety Evaluation

The original plant design often predates formal industry standards, and this is particularly true for the Main Control Board. This Safety Evaluation specifies the use of IEEE Standard 384-1974 "IEEE Standard Criteria for Separation of Class 1E Equipment and Circuits", section 5.6.2 "Control Switchboards". This standard allows that minimum separation distance between redundant Class 1E equipment and wiring internal to control switchboards can be established by analysis of the proposed installation.

In the five as-built cases, the use of technical evaluations determined that we have no unreviewed safety questions. The analysis demonstrates that, in worst case faults, insufficient energy is generated to cause damage before the fault is interrupted by a qualified device. It is further demonstrated that for extended overload cases the continuous current rating of the circuit components exceeds the current melting value of the qualified device.

4. Safety Evaluation 351 - Vessel Injection Motor Valve (MOV) Low Head Basis for Opening

Description of Change

Residual Heat Removal (RHR) to Reactor Vessel (RxV) Injection MOVs normal position was changed from normally closed to normally open. The affected MOVs are MV-32064, 32065, 32167, and 32168.

Summary of Safety Evaluation

The RHR to RxV MOVs are subject to hydraulic locking in the closed position. To prevent this, the valves will be left open during normal operation. These types of flexible wedge gate valves can potentially become locked when experiencing high differential pressure.

To support the opening of these valves, a Safety Evaluation was written. A PRA was completed to form the basis of the evaluation. The PRA analyzed the core damage frequencies associated with maintaining the MOVs normally open and maintaining the MOVs normally closed.

The PRA determined that with the MOVs normally open, the risk of an intersystem LOCA increases slightly. However, with the MOVs normally closed, the probability of a failure to open is higher. The PRA concluded that core damage frequency was higher with the valves normally closed. Using the PRA as the basis, the Safety Evaluation established that it is prudent to open the MOVs.

5. Modification 90L223 - NIS Power Range Meter Upgrade

Description of Change

Analog Nuclear Instrumentation System (NIS) Power Range (PR) meters are replaced with digital meters. There are three meters per channel and there are four channels. Meters on both Units 1 and 2 are replaced.

Summary of Safety Evaluation

The Safety Evaluation establishes the safety-related functions of the NIS PR channels as Reactor Protection System trip signals for various low, high, and transient power conditions. Additionally, the PR channels provide safety-related bypass permissives for low power trips from the source and intermediate range instruments during startup.

The relevant accidents reviewed were the uncontrolled RCCA withdrawal from subcritical and at power and the excessive heat removal due to feedwater system malfunction. This modification has no effect on these analyses.

The modification is a meter indication change only. The seismic review was completed by Westinghouse, the original design organization, and supplier of the new meters. An internal wiring review was also completed. These reviews concluded that the new components would not have an effect on any systems or components and could not create a new or unreviewed safety issue.

6. Modification 90L213 - Replace Flow Transmitters With EQ Type, FT-626 and FT-928

Description of Change

Flow transmitters 1FT-626, 1FT-928, 2FT-626, and 2FT-928 are all replaced with environmentally qualified (EQ) transmitters.

Summary of Safety Evaluation

These transmitters are located in a harsh environment in the Containment Spray Pump room. They need to be EQ qualified due to high radiation exposure during certain accidents, which are specified in R.G. 1.97. The Safety Evaluation verifies that, since this modification only replaces existing equipment with functionally equivalent EQ equipment, there are no safety issues.

7. Modification 92L386 - Valve Stem Packing Leakoff Lines Capping

Description of Change

Two-inch and larger valves in the Safety Injection (SI), Residual Heat Removal (RHR), and Chemical Volume Control (CVC) systems have had their stem packing leakoff lines capped. Post maintenance testing, inservice leak tests, and component and system surveillances have substantially reduced potential for stem leakage.

Summary of Safety Evaluation

The design goal of the valve stem packing leakoffs for 2-inch and larger valves, operating in radioactive fluids at greater than 212 degrees F, was to limit stem leakage as much as possible. However, by capping the leakoff lines, and filling in this gland area with packing, and by using current maintenance standards, lower leakage actually results.

The Safety Evaluation reviewed the impact that the modified system configuration would have on the USAR and Technical Specifications. The evaluation determined that no adverse impact would result and that the performance of these systems would not be adversely affected. Nor were there any new failure modes or unreviewed safety questions identified.

8. Modification 86L898 - Modification of AFW Pump Lube Oil Piping and Lube Oil Cooling Change to Use AFW Pump Discharge Recirc Flow for Lube Oil Cooling

Description of Change

The AFW Pump lube oil system was modified. The lube oil piping on the AFW Pumps was lowered to insure that the lube oil piping stayed full of oil while the pumps are in Standby. This decreases the probability of air binding of the shaft-driven lube oil pump.

In addition, the source of lube oil cooling was changed. Lube oil cooling had come from the Cooling Water system, but the modification changed the lube oil cooling to AFW Pump recirc flow. This change eliminates the dependence of lube oil cooling on Cooling Water and was a recommendation from the AFW Pump System Reliability Study, which was done for Generic Issue 124.

Finally, since under unusual conditions it is possible for the lube oil cooler to be pressurized to AFW Pump discharge pressure, the lube oil coolers were replaced with coolers designed for 1700 psig. This prevents water intrusion into the lube oil system even if lube oil cooling pressure rises to AFW Pump discharge pressure.

Summary of Safety Evaluation

The Safety Evaluation concluded that the reliability of the AFW Pumps was increased and that this increased the margin of safety for the health and safety of the general public.

The Safety Evaluation reviewed the lube oil piping and cooler change and upgrade to quality class type I. Lube oil cooling to recirc piping change and upgrade to type I was also evaluated. The modified system was reviewed against all identified failure modes. No new, more severe failure modes were discovered.

9. Safety Evaluation 340 - Containment Fan Coil Unit Damper Control Circuit Classification

Description of Change

Plant as-built configuration has the cabling for local control switches for the fan coil unit dampers in non-trained cable trays. The evaluation determines that this is an acceptable condition.

Summary of Safety Evaluation

The evaluation reviewed the effects and probability of failure given the present cable configuration. The evaluation concluded that there was no credible failure that could prevent the safety-related containment cooling function from occurring when required.

Failure initiating events included fire, HELB in the Auxiliary Building, seismic event, and other external events. The response to the accidents, LOCA and MSLB in containment, for which the containment cooling function is required, is not affected. Also, no new safety issues were identified.

10. Modification 93L398 - Charcoal Filter Bypass With Permanent Piping

Description of Change

ADT Collection Tank Discharge goes from the ADT Cartridge Filters directly to the Charcoal Filter, with no permanent filter bypass capability. The Charcoal Filter is effective for removing organic material. However, there has historically been little or no organics, so the charcoal function is generally unnecessary. This modification installs bypass piping around the filter, but maintains filtration capability, if the need occurs.

Summary of Safety Evaluation

The USAR states that there are no credible accidents in the liquid rad waste system that could harm the public or result in a release in excess of Technical Specification limits. This modification does not change any of the assumptions that lead to that conclusion, nor create any new or unreviewed issues.

11. Modification 83L769 - Waste Gas Compressor Upgrade

Description of Change

Added flexible piping to 121 and 122 Waste Gas Compressors (WGCs) to remove pipe strain. Added seal water strainer blowdown valves to 121, 122, and 123 WGCs. Instrumentation was also upgraded to enhance moisture level control.

Summary of Safety Evaluation

The Safety Evaluation established that all of the applicable design and testing standards in effect at the time of installation were maintained. This includes pressure testing of pressure boundary components, stress analysis, and preoperational and operational testing of the modification. With these controls in place, the conclusion was that no safety analyses in the USAR were affected, nor were any new concerns generated.

12. Modification 92L362 Part E - RC Gas Vent System Orifice Bypass Valve

<u>Description of Change</u>

A bypass line, with isolation valve, was installed around the RC Gas Vent System orifice. The line is to be used when in shutdown to aid reactor draindown activities.

<u>Summary of Safety Evaluation</u>

The valve and line meet all the required design and construction criteria applicable. A failure analysis has concluded that all possible events are bounded by the present accident analysis and that no new, unreviewed accidents are created.

13. Modification 91L259 - Containment Fan Coil Damper Switch Relocation

<u>Description of Change</u>

The modification consisted of adding cams to the fan coil unit damper shafts and adding limit switch mounting structures to the damper assemblies. Limit switches are also relocated and replaced. The purpose of the modification was to improve the reliability of damper position indication.

<u>Summary of Safety Evaluation</u>

The function of the limit switches was not changed. Some hardware, air lines, electrical, needed to be adjusted to accommodate the new limit switches; however, the specifications for that adjusted equipment were not changed. The environmental, seismic, and other accident design specifications for the limit switches were not changed.

The safety evaluation reviewed the design for any new mechanical failures that could result from the modification. For example, binding of the lever arm pivot shaft in its bushing, lever arm breakage, and rotation of the cam on the damper shaft were evaluated. All of these were taken into account during design, and none of the new failure modes were determined to be credible.

14. Safety Evaluation 335 - Bottom Mounted Inst Flux Thimble Wear

<u>Description of Change</u>

NRC issued Bulletin 88-09 addressing thimble thinning in Westinghouse reactors: thinning had been reported at several sites. The Westinghouse Owner's Group was tasked with establishing an inspection program. Until Westinghouse could complete its work, interim inspection criteria were used. Eventually Westinghouse issued WCAP-12866, which formally established the inspection criteria. This safety evaluation implements the new criteria.

Summary of Safety Evaluation

Westinghouse conducted thimble collapse tests to determine a safe, allowable wall thickness for the thimbles. The test data was then used to create a thimble inspection program. This safety evaluation determines that it is safe for Prairie Island to implement the program.

Since leakage as a result of a guillotine break is 5 gpm or less, the inspection program cannot result in an unbounded USAR accident. Additionally, wear loss has been very stable over the last ten years, so implementing the new criteria will not increase the probability of an accident. The thimbles are passive devices and autonomous from other plant systems; therefore, there is no new accident requiring evaluation.

15. Modification 91L337 - Cathodic Protection System Upgrade

Description of Change

Three new deep-well cathodic protection systems, ten new test reference cells, and one new underwater reference cell at the end of the emergency water intake pipe are installed. Platinum-based anodes are installed in 200 ft wells with DC rectifiers supplying power to the wells. This system protects the entire site.

Summary of Safety Evaluation

The cathodic protection system has no direct impact on any systems in the safety analysis. The system is intended to increase the reliability of buried steel structures and piping by protecting them from corrosion. This change improves upon the original system and introduces no new safety concerns. In addition, the safety evaluation concludes that adequate measures will be taken during the construction activity to assure no new unreviewed failure modes are introduced.

16. Modification 92Y170 - Cooling Water Piping Replacement

Description of Change

The cooling water supply header was replaced with 1/2 inch wall piping versus the 3/8 inch wall original. In addition, a protective epoxy coating was installed on the inside diameter of the main header. Also replaced were the first isolation valves in each of the supply header branch connections. Isolation valves were added to the main header and non-essential check valves were removed. The cooling water supplies to the AFW Pump were separated, and the alternate cooling supply lines to the Unit 1 Diesel Generators were also removed.

Summary of Safety Evaluation

The safety evaluation determined that all the original codes of construction were satisfied, along with stress analyses for the increased wall thickness. The application and usage of the epoxy coating on the inside diameter of the piping was reviewed to assure that the failure of the coating would not result in system blockages. Detailed hydraulic analysis was accomplished to assure all specifications and USAR assumptions are satisfied.

The safety evaluation also reviewed the installation plan. Specifically the plan to maintain core cooling at all times was reviewed to assure that the required equipment was available. The safety evaluation concluded that there are no unreviewed safety questions, and the modification is bounded by the USAR.

17. Safety Evaluation 358 - Removal of Post LOCA Hydrogen Control Motor Valves from GL 89-10 Program

Description of Change

The Post LOCA Hydrogen Control MOVs were permanently removed from service and subsequently taken out of the GL 89-10 program. An additional result was that the reference to the capability of hydrogen control through the use of containment repressurization was deleted from the USAR. The affected valves are MV-32274, MV-32276, MV-32293, and MV-32295.

Summary of Safety Evaluation

The safety evaluation demonstrated that the only safety related function of the Post LOCA Hydrogen Control MOVs was to provide a containment pressure boundary. The installation of the Containment Hydrogen Recombiners for containment hydrogen control made the Post LOCA containment pressurization system obsolete as a method of hydrogen control. The safety evaluation further established that the MOVs satisfy the requirements of GL 89-10 for their removal from the 89-10 program.

18. Modification 89L165 - U1/2 Component Cooling Surge Tank Crosstie Line

Description of Change

The surge tanks of Unit 1's and Unit 2's Component Cooling Systems are crosstied using a 1-1/2 inch isolable line. The line equalizes the level and pressure in the tanks. This prevents slight leakage across Component

Cooling System interface valves from causing level buildup and level decreases in the respective surge tanks.

Summary of Safety Evaluation

The interactions between the two Component Cooling Systems due to system cross leakage and due to cross connecting the surge tanks was evaluated. No new safety concerns were identified. The system design bases were not affected, and this includes surge tank and train isolation capabilities.

Also evaluated was the failure of the cross connecting piping. The line is isolable, and failure will not disable the Component Cooling Systems. The safety evaluation concludes that the modification does not impair the ability of the Component Cooling Systems to perform their safety functions nor create any new, unbounded failure modes.

19. Modification 90Y095 - Technical Support Center (TSC) Upgrade HVAC

Description of Change

The TSC boundary is extended by modifying the Admin Building Annex upper level to meet the requirements of NUREG 0737. The HVAC system is modified so that make-up and recirculated air going to the upper level can now be routed through the TSC cleanup unit. Also, a new higher capacity AC unit replaced the previous unit along with upgrading humidification capabilities. Finally, an accountability card reader was installed for the upper level as were better air tightness doors.

Summary of Safety Evaluation

The safety evaluation reviewed the proposed changes to the TSC to assure that there were no new unreviewed safety questions. During construction, portions of the TSC did not meet design requirements for TSC activation. However, either the upper or lower level of the TSC remained operational at all times.

In addition, the new system has higher air flow but lower normal return air filtration. This issue is addressed by preserving the capability of manually aligning dampers for increased air volume filtration, if needed. The safety review identified no other items requiring consideration.

20. Modification 93L393 - New Fuel Racks Rearrangement

Description of Change

The new fuel storage racks will be modified to increase storage capacity from 55 to 74 new fuel elements. Originally the new fuel racks could accommodate 88 elements, but 33 storage locations were later removed from service. This modification will return to service these 33 locations while removing from service 14 other locations.

Summary of Safety Evaluation

There were no accidents in the USAR that are affected by this modification to the New Fuel Storage Pit. However, a USAR design basis is the prevention of inadvertent cricicality in the new fuel pool.

A Westinghouse criticality analysis has specified the allowable new fuel arrangement. The safety evaluation then identifies the two critical issues as either placing elements in unallowed locations or a structural failure of the pool. However, the modification will physically prevent misplacement of new fuel elements, and no new load-bearing components will be changed. With these determinations, the safety evaluation concludes that there is no safety issue.

21. Safety Evaluation 328 - D1 Gen Tornado Missile Hazard Protection

Description of Change

This evaluation identifies and documents the Standard Review Plan Section 3.5 (NUREG 0800) as one of the appropriate means for determining tornado generated missile protection requirements. This is a requirement of CDC4.

Summary of Safety Evaluation

The safety evaluation determines that using the criteria specified in NUREG 0800 (SRP) results in the conclusion that tornado protection for the D1 entrance door is not required. The probability that missiles generated by natural phenomena will impact the D1 door and contribute to a 10CFR100 exposure event is below the 10E-07 per year criteria specified in the SRP. Because of the low event probability and per NUREG 0800, the existing as-built D1 entrance is consistent with GDC4.

The key assumptions used in the PRA were the use of EPRI NP-768 and 769 "Tornado Missile Risk Analysis". No credit was taken for the Service Building exterior walls or intervening components. Only Unit 1 is considered, since D5/D6 are dedicated to Unit 2.

22. Modification 92L368 - Unit 2 Cycle 16 Core Reload

Description of Change

This modification replaced depleted Unit 2 fuel assemblies with a fresh reload of 48 Westinghouse VANTAGE + fuel assemblies allowing another cycle of power operation. The new fuel assemblies are enriched to a nominal 4.95 w/o U235 and results in a projected cycle length of 19,960 MWD/MTU, which includes a 21-day coast to approximately 79% of full power. This is equivalent to 518 effective full power days.

Summary of Safety Evaluation

The Unit 2 Cycle 16 reload was developed by the NSP Nuclear Analysis Department using methodology addressed in NSPNAD-8101-A, Qualifications of Reactor Physics Methods for Application to PI Units.

The following safety concerns were addressed in the safety evaluation:

A. Thermal Hydraulic Analysis
B. Accident and Transient Analysis
C. LOCA-ECCS Analysis
D. Rod Ejection Analysis
E. Fuel Handling Accident
F. Refueling Shutdown Margin
G. Heatup/Cooldown Curves - Reactor Vessel Radiation Surveillance Program
H. Fuel Rod Design Performance
I. Spent Fuel Heat Load
J. New Fuel Rack/Spent Fuel Rack Criticality
K. Core Exposure Limits/Off-site Dose Calculations
L. Fuel Assembly Design Changes
M. Startup and Operation
N. Validity of Safety Evaluation

All results were acceptable and are presented in NSPNAD-93007, Rev. 0, Prairie Island Unit 2 Cycle 16 FRDR, SOR and RSE Report. The LOCA analysis was performed by Westinghouse and is documented in the Unit 2 Cycle 16 LOCA Confirmation Letter 93NS*-G-0051, November 30, 1993. This letter confirms that the operation of Prairie Island Unit 2 Cycle 16 will continue to conform to the acceptance criteria of 10CFR50.46.

In conclusion, since all transient analysis met the acceptance criteria, there are no unreviewed safety questions for the PI Unit 2 Cycle 16 Core Reload Modification.

23. Modification 94L437 - Unit 1 Cycle 17 Core Reload

Description of Change

This modification replaced depleted Unit 1 fuel assemblies with a fresh reload of 48 Westinghouse VANTAGE + fuel assemblies allowing another cycle of power operation. The new fuel assemblies are enriched to a nominal 4.95 w/o U235 and results in a projected cycle length of 21,167 MWD/MTU, which includes a 26 day coast to approximately 74% of full power. This is equivalent to 550 effective full power days.

Summary of Safety Evaluation

The Unit 1 Cycle 17 reload was developed by the NSP Nuclear Analysis Department using methodology addressed in NSPNAD-8101-A, Qualifications of Reactor Physics Methods for Application to PI Units.

The following safety concerns were addressed in the safety evaluation:

A. Thermal Hydraulic Analysis
B. Accident and Transient Analysis
C. LOCA-ECCS Analysis
D. Rod Ejection Analysis
E. Fuel Handling Accident
F. Refueling Shutdown Margin
G. Heatup/Cooldown Curves - Reactor Vessel Radiation Surveillance Program
H. Fuel Rod Design Performance
I. Spent Fuel Heat Load
J. New Fuel Rack/Spent Fuel Rack Criticality
K. Core Exposure Limits/Off-site Dose Calculations
L. Fuel Assembly Design Changes
M. Startup and Operations
N. Validity of Safety Evaluation

All results were acceptable and are presented in NSPNAD-93004, Rev. 0, Prairie Island Unit 1 Cycle 17 Final Reload Design Report. The analysis for containment response to a steam line break was found to have some nonconservatisms in it. Through discussions with the NRC, it was determined that a JCO was appropriate to allow operation until new methodology demonstrating containment response acceptability can be approved. The LOCA analysis was performed by Westinghouse and is documented in the Unit 1 Cycle 17 LOCA Confirmation Letter 94NS*-G-0011, April 20, 1994. This letter confirms that the operation of Prairie Island Unit 1 Cycle 17 will continue to conform to the acceptance criteria of 10CFR50.46.

In conclusion, since all transient analyses met the acceptance criteria, there are no unreviewed safety questions for the PI Unit 1 Cycle 17 Core Reload Modification.

24. Safety Evaluation 382, Revise USAR to Discuss Change to Subcooling Margin Monitor Range

Description of Change

Configuration Management-Related Follow-On Item (FOI) No. A0471, "Subcooling Variable Range," noted that the indicated Subcooling Margin Monitor (SMM) range has been changed. In response to NRC Generic Letter 82-33, Prairie Island indicated to the NRC that the indicated SMM range was 999°F subcooling to 200°F superheat. The assessment of FOI Nol A0841 notes that the SMM range was changed to 200°F subcooling to 35°F superheat when the SMM meters on the control board were replaced.

Summary of Safety Evaluation

This safety evaluation justifies the change to SMM range. The current range is consistent with the range recommended by the NRC. The range recommended by Reg Guide 1.97 is the same as the installed instrumentation. The current range does not affect the operability of any of the Inadequate Core Cooling Monitor System nor any components within the system.

25. Modification 90Y115, Unit 1 - Pressurizer Surge Line Whip Restraint Modification

Description of Change

This modification removed shim packs from pressurizer surge line whip restraints to satisfy ASME stress limits and NRC Bulletin 88-11. These whip restraints are no longer necessary following the elimination of surge line rupture as the structural design basis for Prairie Island Unit 1.

Summary of Safety Evaluation

Prairie Island USAR Section 4.6.2.2 "Blowdown Jet Forces & Pipe Whip" was reviewed for this modification. However, the NRC has approved the leak-before-break analysis for the surge line which concluded that the probability of large pipe breaks occurring in the pressurizer surge line is sufficiently low such that the dynamic effects associated with postulated pipe breaks need not be a design basis. Pipe whip restraints are no longer a requirement on the Prairie Island Unit 1 pressurizer surge line.

This modification will maintain acceptable margins of safety as evaluated in the USAR and in Technical Specification, and does not involve any unanalyzed safety questions.

26. Safety Evaluation 346, Addition of Titanium Compounds to the Secondary System as Inhibitors of Secondary Side Stress Corrosion Cracking and Intergrannular Attack of the Alloy 600 Steam Generator Tubing

Description of Change

The purpose of this test was to determine the feasibility of continuous use of titanium compounds in the Prairie Island steam generators to inhibit the initiation and growth of secondary side stress corrosion cracking of the mill-annealed Alloy 600 steam generator tubes. Items for investigation;

1. Ability to add the titanium compound to the feedwater system.
2. Ability to analyze for titanium in the feedwater, steam generator, main steam, and heater drain systems.
3. Amount of the titanium compound to add to the feedwater in order to obtain 5 to 10 ppb titanium in the steam generator blowdown.

Summary of Safety Evaluation

Intergrannular attack/stress corrosion cracking (IGA/SCC) of Alloy 600 tubing at tubesheet and tube support plate locations in the secondary side of steam generators continues to be a major cause of tube degradation and steam generator replacement. With the exception of significant primary side temperature reductions, no remedial measure has been found effective in all cases to stop or inhibit IGA/SCC. (The effectiveness of boric acid addition remains controversial in the industry.)

At Prairie Island this cracking is occurring in the tubesheet crevice region of Steam Generator 12 and has the potential to occur in all steam generators both at the tubesheet crevice region and at the tube support plates. Secondary side stress corrosion cracking at tube support plates is occurring in similar steam generators at Farley Units 1 and 2.

Laboratory research indicates that several surface film forming substances can reduce or stop IGA/SCC in caustic environments. Titanium dioxide is among the substances tested and found to be effective.

The addition of these titanium compounds will decrease the degradation of steam generator tubes from stress corrosion cracking and thus decrease the probability of a steam generator tube rupture. Since the amounts of titanium will be in the ppb range in the steam generator and less in the main steam and feedwater lines, there will be no effect on the Excessive Heat Removal Due to feedwater System Malfunction, the Loss of Normal Feedwater, and the Rupture of a Steam Pipe accidents.

The additional particulate matter is less than that observed in early plant life and thus, for the short period of high injection rates, will not affect the amount of sludge/tube deposits in the steam generator.

27. Modification 90Y100, Auxiliary Building Crane Upgrade

Description of Change

The existing trolley on the auxiliary building crane was replaced with a new trolley which complies with the requirements of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", and NUREG-0612., "Control of Heavy Loads at Nuclear Power Plants". This modification was required for the movement of the TN-40 spent fuel storage casks into the spent fuel pool.

Summary of Safety Evaluation

The probability of a heavy load drop in the spent fuel pool or in the safe load path while handling the cask is being significantly reduced by the upgrade of the auxiliary building crane to a single-failure-proof design in accordance with the requirements of NUREG-0612. The single-failure-proof handling system to be used in cask handling will be in compliance with all applicable regulatory requirements and will provide a significantly more reliable system for handling heavy loads than the original auxiliary building crane could provide.

The use of a single-failure-proof handling system will essentially eliminate the possibility of a heavy load drop accident. This conclusion is supported by the guidance in Section 5 of NUREG-0612 which specifically allows the use of a single-failure-proof handling system in place of the ability to withstand a load drop accident.

28. Safety Evaluation 332, Surveillance Capsule W Insertion

Description of Change

This safety evaluation covers the insertion of surveillance capsule W into the Prairie Island Unit 1 reactor vessel. Surveillance capsule W contains specimens from a Monticello reactor vessel surveillance capsule. These re-encapsulated test specimens are being inserted into the Prairie Island Unit 1 reactor vessel for the purpose of irradiating the test specimens to a fluence corresponding to the end-of-life fluence for the Monticello reactor vessel.

Summary of Safety Evaluation

Surveillance capsule W is identical to the capsules which were originally installed in Unit 1. The only difference is the source of the metal inside the capsule. The capsules weigh approximately 55 pounds with capsule W within 5 pounds in weight of the original Prairie Island capsules. Capsule sealing and installation is also identical.

The present surveillance capsule program allows anywhere from one to six capsules to be installed in the reactor with no fixed order for removal. The installation of capsule W is therefore within the analyzed possibilities as foreseen in the USAR. Since this capsule configuration was anticipated in the USAR, it is within the accident analysis as described in the USAR. The installation of capsule W has no effect on the Prairie Island surveillance capsule program.

All accidents were reviewed and it was determined that installation of this capsule did not affect them. Since the weight of the capsule and its associated handling tool is less than that of a fuel assembly, this is not a heavy load and does not affect the fuel handling accident. The evolution of moving a capsule is within the originally expected scope of activities at the plant and as such does not constitute a new test or evolution.

29. Modification 89Y055, Spent Fuel Bridge Crane Replacement

Description of Change

The original spent fuel bridge crane was replaced with a new crane. The new spent fuel bridge crane has 2 hoists. One hoist is a 2-ton capacity hoist used for general fuel handling and Unit 1 refuelings. The other hoist is a redundant, 3-ton design rated load, 3700 pound maximum critical load hoist used for general fuel handling, Unit 2 refuelings and heavy load lifting. The 3-ton hoist is specifically designed for moving the heavy loads contained in the spent fuel pool enclosure and has the capability to transfer out to the New Fuel Crane outside the spent fuel pool enclosure for movement of new fuel assemblies from the new fuel containers to the new fuel pit.

Summary of Safety Evaluation

The safety evaluation for the replacement of the spent fuel bridge crane included the evaluation of each of the following areas of safety concern: new crane installation, old crane removal, fuel handling with the new crane, heavy load handling, structural calculations, new fuel crane structural improvements and an electrical review. The evaluation of these safety concerns and their resolution formed the basis for concluding that the spent fuel bridge crane replacement did not constitute an unreviewed safety question.

In addition, the new crane and the 3-ton hoist were designed to meet the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants".

30. Modification 91L310, Control Room Monitoring of DC System

Description of Change

This modification resulted from a commitment to the response to Generic Letter 91-06, Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies" which outlines suggested parameters for DC monitoring. It also satisfied concerns addressed by Configuration

Management items FOI A0210 and FOI A0200. It provided the following alarms and analog indications in the Control Room:

Alarms for each of the 125 VDC systems;

1) Battery charger AC input breaker open
2) Battery charger DC output breaker open
3) DC bus overvoltage
4) Battery discharge

Analog indication for each of the 125 VDC systems:

5) Battery float current
6) Battery circuit output current
7) Battery discharge
8) DC bus voltage
9) Time remaining to battery discharge

Alarm for FOI A0210:

10) Battery terminal voltage approaching 105 VDC

Analog indication for FOI A0200:

11) Ambient air temperature of 11, 12, 21, and 22 Battery Rooms

Summary of Safety Evaluation

The safety evaluation addressed the following issues and possible hazards associated with the design and construction implementation of the Control Room Monitoring of DC Systems Monitoring:

1) Capability of the system to perform in accordance with the original and modified design requirements after construction completion

2) Effects of construction on the operating units

3) Effects of testing on the operating units

The specific changes incorporated by the modification enhance Operator interface and do not directly alter system function. In this manner, no new failure modes or unbounded accident analysis scenarios have been introduced by this modification.

31. Safety Evaluation 354, Removal of Safety Injection MOV's MV32068, MV32070, MV32073, MV32171, MV32173, and MV32176 from the Generic Letter 89-10 Program

Description of Change

This evaluation justified the removal of the following valves from the GL 89-10 MOV testing program:

MV32068 SI to Loop B Cold Leg Unit 1
MV32070 SI to Loop A Cold Leg Unit 1
MV32073 SI to Cold Leg Isolation Unit 1
MV32171 SI to Loop B Cold Leg Unit 2
MV32173 SI to Loop A Cold Leg Unit 2
MV32176 SI to Cold Leg Isolation Unit 2

Summary of Safety Evaluation

The bases for removal of these valves is that they are not mispositionable. Technical Specifications require these valves to be locked in the open position. These six valves meet the requirement that the MOV is not intended to change positions at any during a design-basis event or in the plant emergency operating procedures and inadvertent operation from the control room is prevented.

32. Safety Evaluation 383, USAR Change: Section 6.2.3.5 Single Failure Analysis

Description of Change

The purpose of this safety evaluation is provide the justification for a change to the Updated Safety Analysis Report (USAR) regarding a single failure analysis discussion involving Safety Injection Pump Discharge Crossover Line Valves.

The single failure analysis describes how the failure of any single active component will not prevent fulfilling the design function of the safety injection and residual heat removal systems. As presently written, a rupture anywhere in a flow path considered in the USAR evaluation would have to be isolated, and an alternate flowpath used. USAR Table 6.2-8 evaluates the alternate flowpaths to be established, and includes valves SI-14-1 and SI-14-2. The USAR also states that the isolations and alternative flowpaths can be accomplished from the control room. That statement is misleading, because there are manual valves in a potentially high radiation environment during recirculation.

Summary of Safety Evaluation

A technical evaluation of the functional requirements for the SI valves to determine whether the isolation of a ruptured line is required for effective response during an accident. This evaluation is bounded by three scenarios: active failures during injection, active failures during recirculation, and passive failures during recirculation. The argument of this evaluation is that if the SI valves are open during the high head recirculation phase of accident response and a rupture occurs, closure of the valves is not required if the ECCS can provide adequate core cooling without isolation of the valves. The evaluation considers potential failure scenarios, and concludes that valves SI-14-1 and SI-14-2 can be removed from the USAR consideration of the required isolations for single failure analysis.

33. Safety Evaluation 322, Addendum 1, Component Cooling Containment Isolation Valves

Description of Change

Fire Protection Safe Shutdown Analysis indicates that the motor control center breakers for Component Cooling MOVs for the Reactor Coolant Pumps are blocked and tagged open for the purpose of:

> "to ensure maintenance of reactor coolant inventory using the Safety Injection system (B-Division operation), component cooling water is required to protect the reactor coolant pump seals from damage"

In addition, the Updated Safety Analysis Report (USAR), Table 5.2-1 indicates the MOVs in question can be remotely operated for containment isolation. This is only necessary in the event of a component cooling piping (associated with a reactor coolant pump) leak during a Loss of Coolant Accident. However, as discussed above, the power supply for these valves is removed.

The purpose of this safety evaluation is to address the remote operability requirements of these MOVs as presented in the USAR, Section 5 and to ensure that maintaining the power supply breakers open is acceptable.

Summary of Safety Evaluation

Per the plant's Final Safety Analysis Report, the system satisfies the Atomic Energy Commission General Design Criteria (the criteria applicable at the time of plant licensing) because the component cooling system is a closed system and isolation valves are provided outside containment.

A technical evaluation considered various factors which justify the current arrangement.

> Several alarms and indications are available to alert operators of a rupture in the component cooling piping to/from the reactor coolant pumps. C14 Abnormal Operating Procedures directs operators to locally shut the MOVs (supply and return) in question, minimizing the component cooling system leakage into containment in the event of a leak. The isolation valves are covered by a water seal established by the discharge head of the component cooling pumps and the level of water the component surge tank. In addition, downstream check valves and the

upstream header MOVs assist in limiting the leakage into containment until the MOVs can be locally shut.

Regarding susceptibility to a passive failure, the passive failure is a component cooling piping leak of not more than 50 gpm. This leak, by definition of (NRC SECY-77-439), would occur at least 24 hours after the event. After 24 hours, containment pressure is less than 5 psig. Thus, component cooling system supply and return header pressure are greater than containment pressure when a component cooling fluid leak is postulated, and any leakage is into containment. Therefore, local closure of the MOVs (in conjunction with the water seal) would prevent post-accident containment atmosphere from escaping. The maximum postulated leak of 50 gpm allows time for operator action to maintain surge tank level and isolate the leak.

34. Safety Evaluation 97, Addendum 7, Reactor Coolant Oil Pan Aluminum Evaluation for Containment Hydrogen Production

<u>Description of Change</u>

This evaluation was performed to determine the hydrogen production rates in containment due to aluminum parts on the reactor coolant pump motors that were not previously identified. Furthermore, it provides the time it takes to reach 3.5% hydrogen concentration in containment.

<u>Summary of Safety Evaluation</u>

The evaluation concludes that there is adequate operator response time to place the containment hydrogen recombiners in service.

35. Safety Evaluation 360, ANSI B31.1-1967 to ASME B31.1-1989 Code Reconciliation for Fabrication and Erection

<u>Description of Change</u>

This safety evaluation reviews and documents the safety significance of updating the Prairie Island fabrication and erection code from ANSI B31.1-1967 to ASME B31.1-1989. It should be noted that only the Code edition will be changed, other fabrication, erection, inspection, and examination items from the original X-Hiawatha 106 Specification which go beyond the Code remain the same. This applies to both safety related and non-safety related piping.

<u>Summary of Safety Evaluation</u>

A complete Code reconciliation was performed. The reconciliation report concludes that there were no significant technical changes to the Code that affect fabrication or erection requirements. Furthermore, changing the fabrication and erection Code has no impact on the design allowable stresses nor on the design input assumptions. The majority of the changes were either editorial or administrative.

36. Safety Evaluation 361, Refueling Water Storage Tank Boron Concentration Evaluation

Description of Change

This safety evaluation documents the acceptability of increasing boron concentration in the Refueling Water Storage Tank (RWST). With the use of higher enriched fuel, a greater RWST boric acid concentration is necessary to maintain adequate shutdown margin. A higher RWST boric acid concentration results in a lower spray and sump pH during accident mitigation. This evaluation reviews the resultant effects of reducing the containment spray and sump pH during accident mitigation.

Summary of Safety Evaluation

Calculations indicate the lowest injection (spray) and sump pH are as follows (these are based on a 3500 ppm boron concentration in the RWST and Accumulator, 4000 gallons in the Boric Acid Storage Tank, and minimum sodium hydroxide concentration in the Caustic Standpipe):

Condition	pH
Injection (Spray)	8.6
Recirculation (Sump)	7.7

Based on these pH results, two concerns needed further evaluation for acceptability:

Effect on off-site dose analysis, and
Potential for stress corrosion cracking of the austenitic RHR piping.

The evaluation shows that there is no effect on the off-site dose analysis results.

The evaluation indicates that there will be no significant increase in the probability for corrosion cracking of the austenitic piping.

The evaluation concluded that the increased RWST boric acid concentration would not inhibit long-term post-LOCA cooling.

Therefore, there is no adverse impact of allowing the sump pH to decrease to 7.0, or allowing the RWST an Accumulator boron concentration to increase to 3500 ppm.

37. Safety Evaluation 342, Place CT-BT112 in MANUAL for All Operating Conditions

Description of Change

An NRC Engineering Evaluation Report AEOD/E90-05, dated July 25, 1990, "Operation Experience on Bus Transfer" analyzes several automatic bus transfer failure events. It incorporates information from EPRI and several recent power industry studies on bus transfers. The report examines four transfer schemes: an in-phase transfer for which Prairie Island has no counterpart, a simultaneous fast transfer similar to the reactor coolant pump transfer, a sequential transfer similar to the circulating water and cooling tower bus transfers, and a dead bus transfer similar to the safeguards bus transfer.

Summary of Safety Evaluation

Comparing the report to the Prairie Island configuration shows one item of concern to safety related equipment. If a safeguards bus is supplied from either of the cooling tower 4.16 kV busses, a fast transfer to CT-BT112, "4.16 kV Bus Tie: CT11 to CT12", could result in a significant angle between the incoming source voltage and the bus residual voltage. There is potential then for damage to safety related equipment connected to the bus.

An Operational Experience Assessment was performed which recommended that the Auto/Manual Switch for breaker CT-BT112 be placed in manual. This would prevent an automatic transfer potentially damaging to safety related equipment. It would also result in the actuation of safeguards bus sequencer in the event of any undervoltage on Bus CT11 or CT12. This safeguards actuation might not occur if CT-BT112 were in "Auto". However the sequencer is designed to minimize potential damage to safety related equipment. The decrease in probability of damage to safety related equipment that results from this damage outweighs the increase in probability of safeguards actuation.

Exhibit B

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Revision 12 to the Updated Safety Analysis Report

Instructions:

1. The List of Effective Pages, the Table of Contents, and Chapters 1 through 10 have been reprinted in their entirety. Please discard the pages, tables and figures in these sections and replace with the new Revision 12 version of these sections.

2. For Chapters 11 through 14 and Appendices I and K, remove individual USAR pages, tables, and figures and replace with the new Revision 12 pages provided in this exhibit AND REMOVE FIGURE 14.4-4. Removed pages should be discarded. Please note the following special instruction:

   a. Remove USAR Appendix 14B, Prairie Island Unit 1 Cycle 15 USAR Update in its entirety and replace with Prairie Island Unit 1 Cycle 17 USAR Update (NSPNAD - 94007, June 1994) included in this exhibit.

   b. Remove USAR Appendix 14C, Prairie Island Unit 2 Cycle 15 USAR Update in its entirety and replace with Prairie Island Unit 2 Cycle 16 USAR Update (NSPNAD - 94002, April 1994) included in this exhibit.

3. Remove Revision 16 of the NSP Operational Quality Assurance Plan (USAR Appendix C) in its entirety and replace with NSP Operational Quality Assurance Plan Revision 17 included in this exhibit.

4. When page removal/replacement is complete, review the USAR Listing of Effective Pages to ensure the copy of the USAR Manual is current and complete. Contact NSP Nuclear Licensing at 612-388-1121, Extension 4662 if you require additional assistance.