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June 30, 1992

10 CFR Part 50
Section 50.71

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Submittal of Revision No. 10 to the
Updated Safety Analysis Report (USAR)

Pursuant to 10 CFR Part 50, Section 50.71(e) we are submitting 13 copies of Revision No. 10 to the Updated Safety Analysis Report (USAR) for the Prairie Island Nuclear Generating Plant. This revision updates the information in the USAR for the period from January 1, 1991 through December 31, 1991.

Exhibit A contains a description and summary of the safety evaluation for changes, tests and experiments made under the provisions of 10 CFR Part 50, Section 50.59 during this period.

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

Included in Exhibit B is Revision 16 to the Northern States Power Company Operational Quality Assurance Plan in compliance with 10 CFR Part 50, Section 50.54(a). Changes in Revision 16 to the Plan are described in Exhibit A (Item 69, page 34) of this letter

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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:

1. 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 10CFR Part 50, Section 50.59(b)

2. A brief discussion of the revision to the Operational Quality Assurance Plan pursuant to 10CFR Part 50, Section 50.54(a)(3)

3. A brief description and a summary of the assessment for each of those changes being made to the USAR as a result of the Design Basis Document Reconstitution program currently being implemented at Prairie Island.

1. Safety Evaluation 323, Reactor Coolant Vent Leak Detection Pressure Transmitter 1PT-729 Drawing Changes

Description

During a walkdown of the Reactor Coolant Gas Vent System, it was noted that the drawings for Leak Detection Pressure Transmitter 1PT-729 did not reflect actual field conditions. The drawings have been updated to reflect the actual physical configuration.

Summary of Safety Evaluation

A safety evaluation was performed to determine that adequacy of the existing configuration. The primary concern evaluated was whether the configuration could lead to an inadvertent opening of the valves that would create a vent path. The evaluation concluded that there were no safety concerns associated with the actual configuration.

2. Safety Evaluation 295, Removing the Fire Protection System from Service for Safeguards Ventilation System PAC Filters.

Description

To work on the fire protection system requires isolating the water supply to three pre-absolute charcoal filters.

Summary of Safety Evaluation

Due to the conservative assumptions in the iodine loading calculation of the charcoal filters and the multiple failures that must occur and that the peak charcoal bed temperature occurs 9 days into an accident, it is concluded that the fire protection water spray system may be out of service for 7 days without creating any additional threat to the health and safety of the public. Administrative controls will ensure that the seven days will not be exceeded.

3. Safety Evaluation 296, EQ Consideration of USAR Section 14 Accidents

Description

Existing Environmental Qualification (EQ) documentation did not address all USAR Section 14 accidents, including rod ejection and dropped rod. In general, the EQ program documentation only covered line breaks. Regulatory Guide 1.89 states that EQ should be considered for equipment needed to mitigate design basis accidents other than a loss-of-coolant-accident (LOCA) or high-energy-line-break.

Summary of Safety Evaluation

Because the LOCA temperature and radiation environments in containment are more limiting than the containment environments for the other non-documented accidents, the equipment relied upon for the accident analysis will not see an environment harsher than the accidents already considered in the EQ program. Therefore, the safety evaluation concluded that equipment needed for the dropped rod and rod ejection accidents and other USAR Section 14 design bases accidents was adequately covered by the existing EQ program.

4. Safety Evaluation 297, Reduced T_{ave} Coastdown Operations

Description

During end of life coastdown operations when the RCS boron concentration reaches 0 ppm, Prairie Island will allow T_{AVE} to decrease to compensate for the loss of core reactivity due to fuel depletion. This allows the unit to maintain 100% power for a longer time and therefore extract more energy from the fuel. One result of this reduced T_{AVE} coastdown operation is that the reactor maintains 100% power at a lower T_{AVE} than is assumed in the plant design and safety analysis.

Summary of Safety Evaluation

Reduced T_{AVE} operation changes the initial conditions assumed in the plant safety analyses. The lower T_{AVE} increases the margin of safety in all the accidents analyzed in the USAR, and does not cause any accident to exceed their acceptance criteria. In addition, all instrumentation and control systems will continue to operate as designed, and will adequately perform their intended safety function.

Exhibit A

5. Safety Evaluation 309, Operation at Increased T_{AVE}

Description

In order to optimize Prairie Island plant efficiency Moisture Carryover testing will be performed in order to take credit for this in our calorimetric program. During testing in order to evaluate the balance of plant and steam generator performance, reactor coolant system average temperature will be varied from 557 to 560 to 562.5 degrees F.

Summary of Safety Evaluation

Plant operation with T_{AVE} up to 562.5 degrees F does not change any of the initial conditions assumed in all the accidents analyzed in the USAR. A higher T_{AVE} does not cause any LOCA or non-LOCA accident to exceed their acceptance criteria. All of the accidents analyzed continue to meet all acceptance criteria with no increase in either offsite or onsite consequences. The increased T_{AVE} operation only changes the initial temperature of the reactor coolant system during an accident. The change in temperature is small and all equipment continues to operate within their design basis. At increase T_{AVE} operation all equipment important to safety continues to operate within their design basis. No equipment is required to operate in any manner not intended in its original design. The plant and all its associated equipment continues to operate as designed, within all current design basis. No new types of operations are involved with increased T_{AVE} operation.

6. Safety Evaluation 314, Emergency Cooling Water Dump Line

Description

The emergency cooling water dump line is still operable with the emergency dump line motor valve closed with its breaker open.

Summary of Safety Evaluation

The closed emergency dump line valve and open breaker protects construction personnel during construction of the D5/D6 project, because the line dumps into the D5/D6 building until the dump line is rerouted. Construction personnel are working in the lower levels of the building and could be trapped and drowned if the dump line were to be operated without notice.

The dump line prevents flooding of equipment important to safety in the turbine building. Even though this equipment is important to operations, it is secondary to personnel safety in the construction area. Analysis concludes that there is 50 minutes to evacuate the construction area, restore the valve breaker and valve line prior to reaching the level at which equipment important to safety would be affected in the turbine building.

7. Safety Evaluation 303, Capping of Containment Penetration 3A

Description

Containment Penetrations 3A, on both Units 1 & 2, were abandoned in 1980. The 3/8" stainless steel tubing was capped with compression fittings. In an audit of the Appendix J leakage rate testing program in June 1991, NRC inspectors required Type B testing of Penetration 3A on Unit 1. The Technical Specifications indicate that no Type B or C testing is required of Penetration 3A.

Summary of Safety Evaluation

The safety evaluation points out that compression fittings are very reliable and very tight. The Type B local leak rate test performed on Penetration 3A (Unit 1) verified this. In addition, calculations showed that even if the compression fittings on both ends of Penetration 3A were removed, the offsite dose would still be considerably less than the dose limits of 10 CFR Part 100.

8. Safety Evaluation 300: Justification for continued Operation-Evaluation of Operability of Unit 2 Reactor Coolant Gas Vent System

Description

This evaluation justified operation of the existing Unit 2 Reactor Coolant Gas Vent system until the scheduled refueling outage in February 1992. (See also Item #35, Modification 91L300)

Summary of Safety Evaluation

Recent analysis has shown that thermal and hydraulic loads during certain post accident operating conditions could cause stresses to exceed normal ASME Code allowables. This evaluation summarizes analysis which demonstrate acceptable operation of the Reactor Coolant Gas Vent System under these conditions.

9. Safety Evaluation 317, Cooling Water Modeling Test

Description

The purpose of the test was to collect cooling water flow and pressure data for an engineering evaluation and validation of the cooling water system computer hydraulic model from United Engineers.

Summary of Safety Evaluation

The design basis for the component cooling heat exchangers and containment fan coil units were reviewed. The test procedure does not interfere with the functions identified in the USAR design basis or Chapter 14 accident analyses. It was determined that the cooling water test would not adversely affect system operability.

10. Safety Evaluation 302, Unit One Reactor Coolant System Narrow Range RTD Bypass Manifold Nonconformance

Description

This Safety Evaluation justified the continued use of the Reactor Coolant System RTD Bypass Manifold in the as found condition.

Summary of Safety Evaluation

During the installation of modification 89L157, replacement of the reactor coolant system narrow range RTDs, nonconformance NCR 79 was documented and resolved through engineering evaluations and analysis. The nonconformance documented that the required depth of insertion for the manifold socket outlets which house the swagelok fittings on the Unit 1 bypass loop manifolds did not meet the originally specified dimension of 17/32 + 1/32 depth. Each of the socket depths were measured and the smallest dimension of the 15 sockets measured was 3/16 inch. An analysis showed that the 1/4" fillet weld used to install the swagelok fittings are stressed to only 75% of their allowable stress for pressure and seismic effects and because of this low stress all code allowables including USAR and ANSI B31.1 primary and secondary stress limits are satisfied.

11. Safety Evaluation 324 Revision 1, 4kV Breaker Redundant Fusing

Description

This safety evaluation justified the continued operation of the plant following discovery of design deficiencies related to the requirements of 10 CFR Part 50, Appendix R. The DC control power circuits for the following 4kV breakers are not completely protected by redundant fusing:

1. 11, 21 Safety Injection Pump
2. 11, 21 Residual Heat Removal Pump
3. 11, 21 Containment Spray Pump
4. 11, 21 Component Cooling Pump
5. 21 Auxiliary Feedwater Pump

Specifically, the lockout relay reset circuit for each of the above breakers is not protected.

Summary of Safety Evaluation

Justification for continued operation until system modifications can be implemented to permanently correct this discrepancy was based on the low probability of a Control Room fire requiring operator evacuation and the compensatory actions taken for the interim period.

1. It is considered very unlikely that a fire in the Control Room would require operator evacuation because:

   a. The fire loading in the Control Room is light.
   b. Equipment is readily available for manual suppression of a fire and prevention of the spread of a fire. The Control Room is provided with fire detectors and manual suppression systems (both portable extinguishers and hose stations outside each door).
   c. The Control Room is continuously manned to provide prompt detection and manual suppression of a fire. In essence, the operators perform fire watch functions in these areas.

2. Compensatory measures taken to mitigate the effects of a Control Room fire:

   a. Replacement fuses and fuse pullers have been staged and clearly identified near the appropriate DC Panels 11 and 21 and in Battery Rooms 11 and 21.

   b. A fire protection procedure has been modified to provide direction to the operators to verify if control power to the 4 kV buses is still available, and replace fuses if necessary. In addition, the procedure instructs the operators to place the pump local/remote switches in LOCAL.

12. Safety Evaluation 305, Shutdown Margin Verification

Description

This evaluation addressed the adequacy of shutdown margin for control rod troubleshooting.

Summary of Safety Evaluation

The safety evaluation calculated the negative reactivity additions required to allow for abnormal control rod motion during testing. The rod control system experienced a failure that resulted in a trip. The subsequent troubleshooting procedure required that the rods be moved out of sequence and pulled to a higher position than is ordinarily analyzed for at hot shutdown conditions. Additions of boric acid were used to ensure proper shutdown margin was maintained.

13. Safety Evaluation 308, Evaluation of the Emergency Diesel Generator and Diesel Driven Cooling Water Pump Fuel Oil Storage and Day Tank Vent Lines

Description

The vent lines provide vent and overpressure protection of the Fuel Oil Storage and Day Tanks. The vent lines are located outside the Turbine and Screenhouse Buildings. The adequacy of this configuration was questioned.

Summary of Safety Evaluation

The evaluation addressed the protection provided for the vent lines and concluded, with procedural revisions, that the present configuration is satisfactory.

14. Safety Evaluation 318, Battery Room Heatup Test

Description

A concern existed regarding the potential Battery Room heatup rate in specific event scenarios. An engineering analysis was performed which predicted that manufacturers' recommended temperatures for the equipment (in the room) could potentially be exceeded. A test was proposed to validate the computer modeling program used in the original room heatup analysis.

Summary of Safety Evaluation

This evaluation provided an engineering evaluation of the suitability for performing a special test of the heatup rate in the Battery Rooms without normal ventilation. This evaluation also outlined the administrative controls to be used for performance of the test.

15. Safety Evaluation 319, Justification for Continued Operation: Battery Room Heatup during Loss of Offsite Power

Description

Concern was raised regarding the capability of the equipment in the Battery Rooms to perform their safety design functions in the event a postulated Loss of Offsite Power event causes loss of ventilation to the rooms and the resulting temperatures exceed the design temperatures of the subject equipment.

Summary of Safety Evaluation

The continued operation of the plant until equipment modifications effect a permanent solution was justified on the basis of three reasons:

1. The equipment is inherently resistant to damage from transients in ambient temperature.

2. The ability to provide timely control of ambient temperatures using non-safety related systems is substantial.

3. The probability of a Loss of Offsite Power of a duration that could potentially damage the Battery Room equipment is small.

16. Safety Evaluation 298, Rev. 1, Containment Fan Coil Unit Cooling Water Return Motor Operated Valve Operability Assessment

Description

The subject of this safety evaluation is the installation of nylon wire nuts on dual voltage valve motor operators that are within the scope of the Prairie Island Nuclear Generating Plant Environmental Qualification program. Containment Fan Coil Unit (FCU) cooling water return motor valves were previously determined to have no post-accident functional requirements and, therefore, did not require environmental qualification. This conclusion was documented in an earlier safety evaluation. However, subsequently the Prairie Island Nuclear Generating Plant Design Bases Document Program identified a post accident functional requirement for these valves that required that these valves be returned to the Prairie Island Nuclear Generating Plant Environmental Qualification program (reference Prairie Island Nuclear Generating Plant LER 90-018). The issue of qualification of the nylon wire nuts on these valves was raised during the NRC inspection on Regulatory Guide 1.97 implementation during the week of April 8, 1992.

Summary of Safety Evaluation

The safety evaluation concluded that the safety of the plant and the general public was not compromised with the installation of the nylon wire nuts in an unqualified status based on the probability of an event occurring which required functionality of these valves, the environmental conditions that will exist when valve operation would be required, testing performed by other utilities on valves in a similar configuration, and the consequences of valve failure. This safety evaluation was reviewed by the NRC inspector who raised the issue during the audit and revision 1 to the safety evaluation reflects the incorporation of his comments and additional concerns.

17. Safety Evaluation 313, Limitorque Actuator Limiter Plate Removal Evaluation

Description

The purpose of this safety evaluation is to review the removal of limiter plates from Limitorque motor actuator torque switches in safety related applications and administratively control the maximum setting of the switch. Installations have been encountered where the output torque or thrust of the actuator does not match the limiter plate installed in the motor-operated valve (MOV).

Summary of Safety Evaluation

Prairie Island will implement an administrative control process in the MOV Program in lieu of using limiter plates. The Limitorque torque switch includes a limiter plate which controls the maximum torque switch setting of the actuator. The maximum torque switch setting considers the following constraints:

1. The actuator's capability to generate output torque based on the motor starting torque, gear ratio, and minimum voltage condition.
2. The actuator output torque rating for the gear ratio utilized in the application
3. The valve allowable load ratings (when provided by the valve manufacturer).
4. The spring pack's performance range (total travel capacity of the spring pack).

Each of the items listed above identifies a specific constraint. The lowest torque value associated with these four constraints becomes the maximum torque for selecting the limiter plate.

Each of the four criteria above are reviewed prior to removal of a limiter plate.

18. Safety Evaluation 294, High Head Safety Injection Performance Evacuation

Description of Change

Previous flow testing and throttle valve positioning for pump runout had not considered instrument inaccuracies.

Summary of Safety Evaluation

It was found that possible deviations in pump performance would not impact safety analysis. Pump testing at higher flows and minimum NPSH showed acceptable pump operation. Therefore the throttle valves are set to prevent pump damage due to runout flow. Instrument inaccuracies will be incorporated in the future high head safety injection pump testing.

19. Safety Evaluation 307, Diesel Generators and Diesel Cooling Water Pumps Fuel Oil Piping Design Issues

Description of Change

Separation issues related to fire protection and protection from missiles generated by tornados or diesel engines were evaluated. Fire hazards to diesel generator fuel oil day tank vent lines, fire and missile hazards to diesel generator fuel oil supply cross-tie line, and diesel cooling water pump fuel oil supplies were addressed in this safety evaluation. Missile hazards to diesel generator fuel oil day tank vent lines and fuel oil storage tank vent lines were addressed in Safety Evaluation No. 308.

Summary of Safety Evaluation

It was concluded that fire will not spread through the small diameter piping to affect the redundant component for piping in the diesel generator rooms. The diesel cooling water pump fuel oil supply piping was reviewed under an NRC Safety Evaluation Report dated September 6, 1979, showing an aggravated exposure fire hazard was no longer a concern for this fire area.

Damage from missiles was shown not to pose a hazard to the piping. Low missile damage probability, planned response to engine generated missiles and abundant fuel oil supplies proved the present design was adequate.

20. Safety Evaluation 312, Basis for No Differential Pressure Testing or Significantly Reduced Differential Pressure of Selected Motor Operated Valves in Response to NRC Generic Letter 89-10

Description

NRC Generic Letter 89-10 recommends demonstrated operability testing of motor operated valves at design differential pressure and/or flow after required torque switch settings are established. The Ggeneric Letter further states that when design basis testing cannot practicably be performed an explanation should be documented describing alternatives to verify the correct switch settings.

The evaluation discusses the different groups of valves which cannot practicably be differential pressure tested due to system configurations, safety concerns, or equipment damage concerns; it also discusses the different groups of valves which cannot practicably be tested to full design differential pressure and are tested at a pressure substantially lower than the specified pressure due to system configurations, safety concerns, or equipment damage concerns.

Summary of Safety Evaluation

This evaluation addresses each of the above groups separately and provides assurance that the valves will operate when called upon.

21. Modification No. 85Y586, Part 5, Human Factors Modification - Control Panels, Category I Vent Doors Annunciator System Revision

Description of Change

The Category I Vent Door annunciator system was provided with a power source separate from the Balance of Plant annunciator system. This allows the Category I Vent Door annunciator system to remain operable if the AC power source to the Balance of Plant annunciator system is lost.

Summary of Safety Evaluation

All portions of the Category I Vent Door annunciator system are non-safety related and the function of the system remains functionally the same. The safety evaluation considered that the system would be unavailable for only a brief period of time during the switchover of power supplies.

22. Modification 90L252, Unit 2 Cycle 15 Core Reload Modification

Description of Change

This modification replaced depleted Unit 2 fuel assemblies with a fresh reload of 52 Westinghouse Optimized Fuel Assemblies allowing another cycle of power operation. The new fuel assemblies are enriched to a nominal 4.2 w/o U235 and results in a projected cycle length of 19,410 MWD/MTU, which includes a 19 day coast to approximately 81% of full power. This is equivalent to 508 effective full power days.

Summary of Safety Evaluation

The Unit 2 Cycle 15 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:

1. Thermal Hydraulic Analysis
2. Accident and Transient Analysis
3. LOCA-ECCS Analysis
4. Rod Ejection Analysis
5. Fuel Handling Accident
6. Refueling Shutdown Margin
7. Heatup/Cooldown Curves - Reactor Vessel Radiation Surveillance Program

8. Fuel Rod Design Performance
9. Spent Fuel Heat Load
10. New Fuel Rack/Spent Fuel Rack Criticality
11. Core Exposure Limits/Off-site Dose Calculations
12. Startup and Operation
13. Validity of Safety Evaluation

All results were acceptable and are presented in NSPNAD-910022P, Rev. 0, Prairie Island Unit 2 Cycle 15 Final Reload Design Report. The LOCA analysis was performed by Westinghouse and is documented in the Unit 2 Cycle 15 LOCA Confirmation Letter 92NS*-G-0007, February 6, 1992. This letter confirms that the operation of Prairie Island Unit 2 Cycle 15 will continue to conform to the acceptance criteria of 10CFR50.46.

23. Modification 90L217, Installation of Design Class III Replacement 2GT Transformer

Description of Change

Unit 2 main generator transformer (2GT) was replaced with a transformer meeting Design Class III rather than Design Class III* requirements.

Summary of Safety Evaluation

Design Class III and III* classifications are for equipment not important to nuclear safety as defined and specified in USAR Section 12.2. Design Class III* components serve only non-nuclear safety functions but, for economic reasons, are designed for seismic loadings. Therefore, the replacement of 2GT with a Design Class III rather than III* transformer is not a nuclear safety concern.

24. Modification 80Y102, Addition of 4" of Concrete to a Partial Area of the Mezzanine Floor of the Auxiliary Building

Description of Change

This modification added an additional 4" of concrete to a partial area of the mezzanine floor of the auxiliary building for radiation shielding purposes. This additional concrete shielding allows access to the auxiliary building during certain accidents.

Summary of Safety Evaluation

The computed stresses were checked against the allowable stresses stipulated in the USAR, ACI-318-63, using the working stress method. It was confirmed that the loads imposed by pipe hangers, cable trays and miscellaneous loads do not exceed 50 psf through summation of the composite loading data.

25. Modification 89L143, New Fuel Pit Racks

Description of Change

The new fuel pit racks were modified. The storage capacity in the new fuel pit has changed from 88 fuel assemblies to 55 fuel assemblies.

Summary of Safety Evaluation

This modification only changed the storage capacity of the New Fuel Pit. The structural integrity of the New Fuel Pit has not been degraded. Westinghouse performed a criticality analysis, entitled "Criticality Analysis of Prairie Island Units 1 and 2 Fuel Racks." This analysis concluded that the K-effective would be less than 0.95, including uncertainties for 55 assemblies of higher enriched uranium (4.27 w/o U-235). Therefore, the margin of safety has not been reduced by the modification of the New Fuel Pit racks.

26. Modification 86L939, Turbine Driven Auxiliary Feedwater Pump Recirculation Valve Logic Change

Description of Change

This modification rewired the 11 and 22 turbine driven auxiliary feedwater pump auxiliary relays to be consistent with the logic diagrams. In addition, the control valves for condensate makeup to the condenser were rewired to be controlled by the AFW pump steam inlet control valve closed position limit switch.

Summary of Safety Evaluation

This modification corrected a discrepancy found between the logic diagrams, operational characteristics and the electrical schematics for the limit switches on the turbine driven inlet steam supply control valves, CV31998 and CV31999. The problem was a delayed opening of the recirculation valve. This was caused by the recirculation valve being controlled by the open limit switch, instead of the closed limit switch, on the steam inlet control valves. By rewiring the control of the recirculation valve to the closed limit switch, the pump's recirculation valve comes open simultaneously with the pump start and reduces the amount of overspeed of the pump during surveillance testing. Since the pump is lined up for full flow during automatic starts, there was no effect on the safety related operation of the pump.

This modification increased the reliability of the turbine driven auxiliary feedwater pumps by reducing the time during which no recirculation flow existed during surveillance testings and also increased the margin to the overspeed trip setpoint. There was no other effect on pressure retaining components. Under safeguard lineup conditions, this effect is negligible. The probability of mechanical failure of the auxiliary feedwater pumps is reduced because this modification causes the lube oil cooling valve to open sooner, the recirculation valve to open sooner and the condensate makeup valve to shut sooner than under the previous configuration.

27. Modification 87Y820, Auxiliary Feedwater Pump Cooling Water Supply Flushing Tees

Description of Change

A 4 inch tee with a 2 inch drain valve was installed upstream of each auxiliary feedwater (AFW) pump cooling water suction supply valve in order to conduct periodic flushes of the cooling water suction supply lines for each auxiliary feedwater pump without removing the auxiliary feedwater pumps from service. A 2½ inch line was also installed to route the flush water to the cooling water return header. A hose is used to connect the drain valve to the flushing return line during the flush.

Summary of Safety Evaluation

The periodic flushing operation became necessary when Asiatic clams were discovered in the cooling water system. The cooling water supplies to each auxiliary feedwater pump are dead legs in the cooling water system. This modification increases the reliability of the AFW pumps by eliminating the requirement to remove an AFW pump from service when conducting the cooling water supply line flush and by providing a safe reliable method to assure the AFW cooling water supply lines are free from debris.

The effect of the piping changes on the cooling water system piping was analyzed by Fluor Daniel. Stress levels in the affected lines are within USAR allowable. Supports for this line, also, are adequate for the increased loading due to this modification.

This modification introduces a valve which, if left open, could compromise the safety related cooling water supply to the auxiliary feedwater pumps. However, from a probability risk assessment perspective, the normal lineup of the flushing valves, which is a closed and capped valve, is effectively the same as a solid pipe

The cooling water supply line additions were analyzed and installed as safety-related.

The flushing water return line is non-safety related.

28. Modification 89L157, Unit 1 Reactor Coolant Bypass Narrow Range RTD Replacement

Description of Change

The modification replaced the Unit 1 Reactor Coolant Narrow Range resistive temperature detectors (RTDs). Sostman, Model No. 11834B RTDs were replaced with RDF, Model No. 21450 RTDs. Although the RTDs are very similar electrically, the structural installation requirements are very different.

Summary of Safety Evaluation

Evaluation of the applicable documents concludes that the new RDF RTDs meet and/or exceed the original design specifications and that no unreviewed safety questions exist.

29. Modification 89L099, 16/26 Inverter Removal

Description of Change

Inverters 16 (Unit 1) and 26 (Unit 2) were removed and their non-safeguards loads were transferred to large service building inverters.

Summary of Safety Evaluation

Both inverters and their loads are not safety-related. However, Inverters 16 and 26 had provided unnecessary loads for the emergency diesel generators and the 12/22 batteries. This modification, therefore, is an enhancement to the safety of the plant.

30. Modification 81Y174 Parts 1 and 2, Safeguards 480 Volt Transfer Switch and Modification 81Y210, Relocation of Motor Starters

Description of Change

These two projects are considered as one due to the very close interdependency of the design. 81Y210 created two new safety related motor control centers (MCCs) in a mild environment and 81Y174 transferred loads to these MCCs, installed a transfer switch scheme such that these MCCs could be powered from either unit for improved reliability and eliminated a sub-fed safety related MCC by relocating the source of power directly to the safeguards 480 volt bus.

Summary of Safety Evaluation

The following considerations were evaluated:

1. adequacy of short circuit protection
2. adequacy of overload protection

3. provisions for safe installation
4. comparisons of interrupting ratings
5. impact on Appendix R analysis

and it was concluded that there are no unreviewed safety questions.

31. Modification 91L303, 13 Fan Coil Unit, Add Vent Valve to Cooling Water Inlet

Description

This modification installed a vent valve on the cooling water supply line to 13 Containment Fan Coil Unit. The valve will be used as a connection point for fan coil hydro testing and will also aid in draining and venting of the fan coil units.

Summary of Safety Evaluation

Seismic loading is reanalyzed on Flour Daniel Pipe Stress Reports PI-233-XIV and PI-233-XV. The results show that impact on piping stresses and pipe support loadings are minimal and that no pipe support requalification is required.

Potential failures would include an inadvertent valve mispositioning and unacceptable pipe stress or pipe loading failures. Several checks are done to avoid valve mispositioning. This valve is normally only used during draining or hydro testing of the Fan Coil Units. Post test valve line up checks are done to verify proper valve position. Additionally, prior to start up this valve will be checked on system checklist C1.1.19-1.

The additional weight of the valve causes a minimal impact on piping stresses and support loading and does not create any new failure modes not previously analyzed (Flour Pipe Stress Reports PI-233-XIV & PI-233-XV).

32. Modification 89L098, Seal Injection to RCP Drain Valve Installation

Description of Change

This modification consists of adding vacuum breakers and associated vent piping to both Unit 1 and Unit 2 Reactor Coolant Drain Tanks (RCDTs). This modification will allow a vacuum formed in the RCDT to be broken. This will prevent the RCDT pumps from cavitating as a result of low suction pressure due to a vacuum in the RCDT during outages when the RCDT is vented to atmosphere.

Summary of Safety Evaluation

Functions of the system were considered (there are no safety related functions nor are there functions which are taken credit for in the Chapter 14 accident analyses) and it was conc ded that this modification does not introduce any unreviewed safety questions.

33. Modification 90L221 Human Factors Modifications of Main Control Panels A & F Part I - F-2 Panel Reconfiguration

Description of Change

To incorporate the recommendations of the Control Room Design Review Committee covering Human F ctors Engineering considerations, the Unit 2 F Panel Section of the Main Control Board is being modified to support a new arrangement for instrumentation and controls associated with:

* Reheater Drains
* Moisture Separator Drains
* Heater Drain Tank
* Heater Drains
* Circulating Water

Scheduled for implementation during the Unit #2 Cycle 15 Outage, F-2 Panel Reconfiguration partially satisfies NUREG-0737 Supplement No. 1 commitment requirements for completion of overall Control Room modifications.

Summary of Safety Evaluation

The Safety Evaluation addressed the following issues and possible hazards associated with the design and construction implementation of F-2 Panel Reconfiguration:

1. Effects on the operation and safe shutdown of the affected unit from an instrumentation and control device perspective during modification package implementation.

2. Proper design specification and materials usage for changes to the Main Control Board.

3. Effects of construction on the operating unit.

4. Capability of the system to perform in accordance with the original and modified design requirements after c truction completion.

5. Proper training requirements identified/implemented prior to modification turnover to operations.

The structural integrity of the Main Control Board has been maintained by ensuring compliance to the original design requirements for a safety-related structure per USAR Section 12.2.1.5. The specific engineering tasks associated with panel reconfiguration 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.

34. Modification 83L769, Waste Gas Compressor Upgrade

Description of Change

This design change modified the piping for 121, 122 and 123 Waste Gas Compressors to improve their operation and capacity. Flexible piping was added to 121 and 122 waste gas compressors to remove piping strain and seal water strainer blowdown valves were added to all three waste gas compressors to effectively remove debris from the seal water supply. In addition, several instrumentation upgrades were performed to 121 and 122 waste gas compressors to enhance moisture level control and calibration of various control parameters.

Summary of Safety Evaluation

This modification was performed in accordance with applicable plant quality assurance requirements in effect at the time of installation. All flexible metal hose assemblies were tested to 1.5 times the design pressure in accordance with Reg. Guide 1.143. Other pressure boundary items were procured and tested according to normal plant quality assurance requirements. Stress analyses were performed to ensure that no structural integrity concerns were compromised. Operational testing and monitoring of the modification indicates that this modification has enhanced the capacity of the waste gas compressors and improved reliability without compromising any concerns as analyzed in the USAR or subsequent submittals.

35. Modification 91L300, Reactor Vessel Head Vent System Support Modification

Description of Change

A three foot section of pipe where the reactor coolant has vent system connects to the pressurizer relief tank line was modified to eliminate possible overstress conditions due to certain loading conditions. The existing reactor coolant gas vent system pipe support configurations were modified to eliminate the possibility of overstress conditions due to certain loading conditions.

Summary of Safety Evaluation

A Westinghouse letter informed us that a potential issue had been identified regarding the reactor coolant gas vent system. Thus, it was decided to have the structural analyses reviewed and any new load cases identified and analyzed for the reactor coolant gas vent system. A review of the structural analyses determined that only one thermal expansion case was considered, and nine more possible thermal expansion and hydraulic loading cases were identified and analyzed.

It was determined that the stress created in the reactor coolant gas vent system piping and pipe supports due to the newly analyzed loads would result in yielding of various system components and permanent deformatiion. To relieve the possibility of these stresses occurring, the pipe and pipe support configuration was modified. The completed modifications do not have any effect on the operation of the reactor coolant gas vent system. The flow path and the design function were not altered. These modifications decrease the chances of a LOCA due to a reactor coolant gas vent system pipe failure.

36. Modification 894065, Waste Liquid Discharge Line Extension

Description of Change

The liquid radwaste and steam generator blowdown discharge point was extended from the head of the circulating water discharge canal to the discharge structure at the end of the canal through a 12 inch polyethylene industrial pipe in the bottom of the canal.

Summary of Safety Evaluation

A calculation was performed by Stone and Webster which determined that the additional 2700 feet of pipe would not affect liquid waste discharge operations. The ability of the liquid discharge to mix with the circulating water discharge was also investigated by Stone and Webster; a diffuser was installed at the end of the discharge canal to ensure that the liquid waste thoroughly mixes with the canal water before being discharged into the Mississippi River.

The discharge canal monitor was bypassed by this modification. All liquid waste discharges are monitored by radiation monitors, which have the ability to terminate flow should the alarm set points be reached. Since all discharge paths entering the waste liquid discharge line are monitored prior to discharge, no unmonitored release will occur. The effluent discharge point remains the same with respect to the site boundary.

37. Modification 90L228, Spent Fuel Pool Storage Rack
Description of Change

A storage cabinet and a ladder hanger, for two ladders were installed at the west end of the spent fuel pit. The closet door for the old closet was remounted near the south stairway in the drop area.

Summary of Safety Evaluation

The storage cabinet was analyzed, designed, and installed in accordance with II/I seismic design criteria due to its proximity to the new fuel pool.

38. Modification 89L155, Unit 1 Accumulator Level Transmitters 11 and 12 Replacement

Description of Change

The Unit 1 accumulator level transmitters were replaced with a more reliable model.

Summary of Safety Evaluation

The new transmitters exceed the original performance specifications.

The accumulator level transmitters signal is not used for any automatic or manual equipment actuation under accident conditions. The failure of the transmitter does not affect actual accumulator operability.

39. Modification 85L858, 4kV ITE Air Circuit Breakers, Install Close Latch Anti-shock Springs

Description of Change

This modification installed close latch anti-shock springs on ITE Air Circuit Breakers as a result of our response to the Institute of Nuclear Power Operations Significant Event Report 75-03. The springs were installed in the breaker operating mechanisms to prevent inadvertent closing of the breaker upon completion of the charging cycle.

Summary of Safety Evaluation

Inadvertent closing of a 4kV source breaker or diesel generator breaker could lead to out of sync paralleling resulting in possible diesel failure. The probability of these events occurring during normal operation is minimal since the breaker charging cycle follows breaker closure.

The potential exists during the breaker rack-in process, since the springs are charged after the breaker is in the connect position.

Therefore, there is a need for the spring in order to prevent inadvertent closure during this process.

40. Modification 83L761 Reactor Coolant Drain Tank Waste Gas Piping Modification

Description of Change

The pressure relief tank to vent header line was crosstied to the reactor coolant drain tank to gas analyzer sample line.

Summary of Safety Evaluation

This modification allows purging both the pressurizer relief tank and the reactor coolant drain tank to the waste gas vent header, therefore the possibility for air-borne contamination is greatly reduced when the reactor coolant drain tanks are opened during an outage.

41. Modification 91L254 DC Control Power Isolation for 12 Diesel Driven Cooling Water Pump

Description of Change

The control power from 125 VDC panel 17 has been rewired to 12 Diesel Driven Cooling Water Pump Control Panel, and a knife switch has been installed inside the Diesel Driven Cooling Water Pump Control Panel.

Summary of Safety Evaluation

The fuse holder are losing tension due to removing/replacing the fuses to perform testing. The addition of the knife switches for isolation would allow the fuses to remain in place while testing, thus eliminating the tension loss problem. The rewiring provides the fuses with tighter holders.

Prior to taking the diesel driven cooling water pump out-of-service, the Limiting Conditions for Operation, Technical Specification 3.3.D were verified.

This change increases the reliability of the circuit as the DC control power fuses/clips will no longer be challenged during frequent isolations. The knife switch is not located where a seismic event could cause it to impact any QA-1 equipment. The knife switch will have a locking bar which will keep it closed during a seismic event.

42. Modification 91L255 DC Control Power Isolation for 22 diesel Driven Cooling Water Pump

Description of Change

The control power from 125 VDC panel 18 has been rewired to 22 Diesel Driven Cooling Water Pump Control Panel, and a knife switch has been installed inside the Diesel Driven Cooling Water Pump Control Panel.

Summary of Safety Evaluation

The fuse holders are losing tension due to removing/replacing the fuses to perform testing. The addition of the knife switches for isolation would allow the fuses to remain in place while testing, thus eliminating the tension loss problem. The rewiring provides the fuses with tighter holders.

Prior to taking the diesel driven cooling water pump out-of-service, the Limiting Conditions for Operation, Technical Specification 3.3.D were verified.

This change increases the reliability of the circuit as the DC control power fuses/clips will no longer be challenged during frequent isolations. The knife switch is not located where a seismic event could cause it to impact any QA-1 equipment. The knife switch will have a locking bar which will keep it closed during a seismic event.

43. Modification 91L272 Terminal Box 1794 Revise Configuration of Wiring

Description of Change

Some wires were relocated to different terminal points inside Terminal Box 1794 in order to conform to the minimum bend radius for the Kapton insulation.

Summary of Safety Evaluation

The control wiring on the 1A steam generator blowdown sample valve was re-terminated onto different terminal points. The valve is used during steam generator blowdown sampling. The failure modes considered were a short circuit between conductors and a short circuit between conductors and ground.

44. Modification 87L004, Parts A and B, Remodeling of Laundry Room for Radiation Protection

Description of Change

The Laundry and Hot Shower Tanks (L&HSTs) were originally installed to separate laundry water for soap removal. The 121 ADT Evaporator would bind with soap foam if this was not done. This evaporator is no longer used. Drain piping previously routed to the L&HSTs has been rerouted to a barrel which has a filter bag suspended in it for large particle collection. This barrel has a drain which directs flow into the floor drain system. In addition, the laundry room was moved and the auxiliary building special ventilation zone boundary was moved from the south wall to the north wall of the old toilet room.

Summary of Safety Evaluation

This modification does not affect plant operations or safety-related equipment. Openings in the auxiliary building special ventilation zone boundary were administratively controlled during the modification. None of the moved walls are load bearing.

45. Modification 89Y010, Chlorine Monitors Logic Change and Safety Evaluation 306, 121 Chlorine Monitors Removed from Service

Description of Change

This modification added two QA I (safety related) chlorine monitors to the existing 122 control room HVAC system. The control room HVAC system isolation logic will be modified to require that two detectors from either train must detect chlorine in excess of the setpoint before causing the control room to isolate from outside air.

Safety Evaluation 306 addressed the removal from service of the 121 control room HVAC system chlorine monitors. Accompanying the removal from service of the monitors, the associated air supply damper was closed.

Summary of Safety Evaluation

The safety evaluation for the modification considered the original design criteria, including the guidance in Reg Guide 1.95 and the applicable single failure criteria, and concluded that the new logic was acceptable. Additionally, the new monitors are identical to the originals and installed in the same manner (including the location of the sample points). The safety evaluation for the indefinite removal of the 121 control room HVAC system considered the normal and accident operation of the system and concluded that, with the associated damper closed, there are no safety functions which are compromised.

46. Modification 89L104, Change CV-31503 and CV-31586 to Manual Operation

Description of Change

The control valve bonnets were replaced with manual bonnets for three valves, now labeled WL-24-1, WL-25-1 and WL-26-1. The control cables, solenoids and all other related electronic components were removed.

These valves are in the flow path of the non-aerated drains sump tank to the CVCS Holdup Tanks and to the waste holdup tank.

Summary of Safety Evaluation

The conversion of the two valves in the flow path of the nonaerated drain sump tank to the CVCS holdup tank to manual improves the reliability of the system by ensuring the proper system valve line up is maintained even if the programmable controller is lost. Prior to the modification, when the programmable controller failed, the operator was required to fail two valves open by shutting off their local air supply and venting the regulator filters. It is important that this system remain reliable.

The valve in the flow path from the nonaerated drains sump tank to the was changed to manual to prevent inadvertent discharge of primary water to the waste holdup tank.

This system is not safety-related. It does not perform any Safe Shutdown functions.

47. Modification 86L961 Rev 1, Reactor Coolant Drain Tank Instrument Signal to Control Room Panel B

Description of Change

Controls have been added to the Reactor Coolant Drain Tank pumps and to the process instrumentation from the Liquid Radwaste Treatment Panel to the associated Control Room Control Board "B".

Summary of Safety Evaluation

This modification simplifies draindown activities for plant operators by permitting Reactor Coolant Drain Tank pump operation from the associated Control Room without the need of an operator stationed at the Reactor Coolant Drain Tank panel.

The pump controls can be manually controlled or automatic. Control switches and recorders were installed in both Unit 1 and Unit 2 Control Rooms. This was a QA III (non-safety related) project.

48. Modification 89L097, Spent Resin Tank Vacuum Breaker

Description of Change

A vacuum breaker was installed in the spent resin tank overflow to the waste holdup tank line at a location close to the spent resin tank. A small section of the 2" piping was removed, a tee installed in the vertical direction, a reducer and 1" pipe installed, and the vacuum breaker screwed to the end of the 1" pipe. The ambient side of the vacuum breaker was piped to a floor drain so that if the vacuum breaker fails, liquid coming from the Spent resin tank will not contaminate a large area.

Summary of Safety Evaluation

The system and piping involved is QA Type 3 (non-safety related). No safety related systems are interfaced by this equipment.

Failure of the newly installed equipment could result, at the worst, in water being discharged into the floor drain. The vent line from the spent resin tank has a screen installed inside the spent resin tank. This will prevent resin from going into the vent line and then into the floor drain.

The vent line is designated as Waste Gas. However, its actual use is that of an overflow. The overflow/vent line goes from the spent resin tank to the waste holdup tank. The waste holdup tank is tied into the Plant Vent System. There is no possibility of hydrogen or other waste gasses being in the overflow/vent line so that no explosion or airborne contamination danger exists if the line were to break.

49. Modification 83Y475 and 83Y480, Appendix R One Hour Fire Barrier

Description of Change

Mod 83Y475 uses Kaowool to wrap instrument and control cable for a one hour barrier and Mod 83Y480 uses a one hour Thermolag system to wrap power cable that would have had to be derated had Kaowool been used.

Summary of Safety Evaluation

10 CFR Part 50 Appendix R requires that all safe shutdown equipment and cabling be analyzed to verify that one design basis fire would not disable both trains. If cabling for redundant trains was not separated by a three hour rated barrier or a minimum of twenty feet with no intervening combustibles, it would either have to be rerouted or be wrapped with a minimum of a one hour barrier. Stone and Webster Engineering performed a derate study which revealed that only instrument and control cable could be wrapped with Kaowool without a derate. Power cable would have to be enclosed in TSI Thermolag 330-1.

50. Modification 89Y982, "C" Panel Interim Annunciator Modification

Description of Change

Replaced the existing annunciator 47024A/B lampbox with a larger unitized lampbox (47024/47524) in preparation for the D5/D6 emergency generator addition, installed two additional annunciation acknowledgement joysticks on "A" panel and unitized the acknowledgement joysticks on "G" panel.

Summary of Safety Evaluation

This box replacement was completed during the 1991 Unit 1 outage. The new "G" panel annunciator lampbox is functionally equivalent to the previous lampbox system. The Emergency Response Computer System was used for temporary annunciation which provided continuous monitoring of the alarms deemed necessary for safe operations during the outage. The balance of plant annunciator system modified is classified as a non-safety related system, but the components installed in the control room were installed as Seismic Class II over I.

51. Modification 88L058, Pressurizer Pressure Transmitter Replacement

Description of Change

Existing pressurizer pressure instrument transmitters for both units have been replaced by this modification. Minor changes to rack wiring to accommodate change to 4-20 mA signal were required.

Summary of Safety Evaluation

Instrument loop accuracy and time response were evaluated to ensure adequate margin to Technical Specifications and USAR analyses.

52. Modification 89L140, Reactor Makeup Storage Tank Overflow Drain Reroute to Turbine Building Sump

Description of Change

The possible contaminated sources for the Reactor Makeup Storage Tank were flanged off and the overflow piping of the Reactor Makeup Storage Tank was rerouted to allow the overflow to drain to either the auxiliary building or turbine building sumps.

Summary of Safety Evaluation

Since the sources of contaminated water were removed, the overflow could be safely redirected to the turbine building sump. The direction of the overflow path is determined by the plant health physicists.

53. Modification 90L191, Valve Mod on the RVLIS Hydraulics

Description of Change

This modification adds isolation valves between Reactor Vessel Level Instrument System (RVLIS) root valves RC-17-2 (hot leg A) and RC-17-4 (hot leg B), and the respective high volume sensor bellows. This modification also provides the documentation for the removal of RVLIS capillary fill valve cap welds as needed.

Summary of Safety Evaluation

This modification maintains seismic and separation requirements. The valves will be qualified at, or above all design requirements of system instrumentation. This modification will be consistent with the original design requirements of the RVLIS and the Reactor Coolant System and will not increase the probability of malfunction of any safety equipment. The isolation valves will be type QA I (safety related) components and will meet the necessary requirements for materials that are part of the reactor coolant system pressure boundary.

54. Modification 90L249 Units 1 and 2 Bus Duct Blower Annunciator

Description of Change

This modification causes a control room alarm when both the preferred and the standby bus duct blower motors are not running. This is accomplished by connecting a normally closed auxiliary contact of each motor controller in series to energize an alarm window. The alarm gives control room operators advance knowledge of isophase bus duct cooling motor failure and a margin of time to prevent temperature design limits from being exceeded through proper action as outlined in the appropriate Alarm Response Guide.

Summary of Safety Evaluation

There are no reactor or generator trip actions associated with the Isophase Bus Duct Cooling control and protection systems. A review of the USAR, Section 8 was done. No USAR changes will be required. The Isophase Bus Duct system has no safety functions and is discussed only as supplying 20kV power from the generator to the main station auxiliary transformers.

55. Modification 90L253 New Fuel Elevator/Reconstitution Basket

Description of Change

This modification expanded Temporary Modification 89T0001 making the reconstitution basket a permanent component in the Fuel Handling system and making the temporary replacement of the existing new fuel elevator basket with the reconstitution basket a permanent option. The new fuel elevator reconstitution basket is part of the Westinghouse tooling for reconstitution of its removable top nozzle fuel. Its purpose is to provide rigid support for a fuel assembly during fuel repair and to accept any removable top nozzle tooling required for the job. The basket is designed to temporarily replace the existing basket on the new fuel elevator.

Summary of Safety Evaluation

The new fuel elevator reconstitution basket differs from the existing basket in the following ways. The new basket is heavier than the old basket and the new basket has slightly different dimensions than the old basket. The increased weight was shown to be well within the limits of the new fuel elevator hoist and cable. The change in dimensions was verified to not cause any interference both by drawings and by utilizing a diver to check the area. An additional up limit switch was installed to provide redundancy equivalent to the spent fuel crane hoists.

56. Modification 89L087 Rev 1 Emergency Stop on New Fuel Elevator

Description of Change

The purpose of this modification is to install an emergency stop switch near the New Fuel Elevator (NFE) which can be manually opened to interrupt power to the NFE. This can then be used as an independent method of stopping travel of the NFE when it is being used to inspect, repair, or reconstitute irradiated fuel.

Summary of Safety Evaluation

This safety switch is intended to provide a method to interrupt power and is not intended to provide overcurrent protection. The switch is rated at 30 amps, 6000 volts and is equipped with arc shuts. It can therefore be opened safely under the anticipated load which is limited by the upstream breaker to 15 amps.

57. Modification 91L257 Unit 1 Cycle 15 Core Reload

Description of Change

This modification replaced depleted Unit 1 fuel assemblies with a fresh reload of 48 Westinghouse High Burnup Optimized Fuel Assemblies allowing another cycle of power operation. Half of the new fuel assemblies (24) are enriched to a nominal 4.[illegible] w/o U235 with the remaining 24 new assemblies enriched to a nominal 4.2 w/o U235. The cycle length is projected to be 17590 MWD/MTU, which includes a [illegible]9 day coast to approximately 31[illegible] of full power. This is equivalent to 406 effective full power days.

Summary of Safety Evaluation

The Unit 1 Cycle 15 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. More details on the operational parameters can be found in NSPNAD-91015, Rev. 1, Prairie Island Unit 1 Cycle 15 Startup and Operation Report, June 1991.

Exhibit A

The following safety concerns were addressed in the safety evaluation:

1. Thermal Hydraulic Analysis
2. Accident and Transient Analysis
3. LOCA-ECCS Analysis
4. Rod Ejection Analysis
5. Fuel Handling Accident
6. Refueling Shutdown Margin
7. Heatup/Cooldown Curves - Reactor Vessel Radiation Surveillance Program
8. Fuel Rod Design Performance
9. Spent Fuel Heat Load
10. New Fuel Rack/Spent Fuel Rack Criticality
11. Core Exposure Limits/Off-site Dose Calculations
12. Startup and Operation
13. Validity of Safety Evaluation

All results were acceptable and are presented in NSPNAD-91006P, Rev. 0, Prairie Island Unit 1 Cycle 15 Final Reload Design Report. The LOCA Analysis was performed by Westinghouse and is documented in the Unit 1 Cycle 15 LOCA Confirmation Letter 91NS*-G-0036, June 21, 1991. This letter confirms that the operation of Prairie Island Unit 1 Cycle 15 will continue to conform to the acceptance criteria of 10CFR50.46.

58. Modification 89L151, Containment Spray Penetration Test Connection and Recirculation Line Addition

Description of Change

Modification 89L151 installed a test connection for leakage rate testing of the Containment Spray penetrations and additional recirculatiion capacity on each of the Containment Spray Pumps. The recirculation line was added to prevent pump damage due to insufficient flow for cooling during surveillance testing.

Summary of Safety Evaluation

The changes made in this mod enhanced the operation of the Containment Spray Pumps during surveillance testing and also brought the plant into full compliance with 10 CFR Part 50 Appendix J for containment leakage rate testing. The piping was analyzed to assure compliance with the USAR. A failure mode analysis was performed to verify no new failure mechanisms were induced.

59. Modification 90L177, Containment Spray Pump Mini Flow Removal

Description of Change

The 3/4" mini-flow recirculation line on each of the Containment Spray Pumps was removed because it is no longer used. A 2" line installed under Mod 89L151 is now the preferred recirculation path.

Summary of Safety Evaluation

Removal of an unused line in the Containment Spray system simplifies the system and provides fewer components that could potentially fail, hence reducing the risk.

60. Modification 89L084, Condenser Steam Dump Bypass Control Valve

Description of Change

Modification 89L084 removed the condenser steam dump bypass control valves (CV-31101 and CV-31119) from service. The objective of the modification was to eliminate out-of-service control board equipment:

1. manual control stations (43013 and 43513),
2. red and green light indication (44031 and 44531) and create more space for any future needs.

The scope of the project included determinating and abandoning signal cables in the relay room and at the control valve electropneumatic converter (I/P). The air supply to the I/P was removed along with the I/P (the air solenoids, valve positioner, and associated pneumatic tubing remained intact).

Changes to the Control Board and operations procedures were performed within the scope of the feedwater upgrade (87Y785).

Summary of Safety Evaluation

The completion of this project eliminated control board equipment which had no use in any normal of emergency evolutions. No credit was taken for the condenser steam dump bypass control valve in the design basis of the control system.

61. Modification 89Y945, Enhanced Emergency Lighting for Emergency Operating Procedure

Description of Change

As a result of a 1988 NRC Safety Inspection, the adequacy of emergency lighting for operators to perform required tasks defined in Emergency Operating Procedure ECA 0-0 during the loss of all AC power was questioned. Operations and engineering personnel walked down the procedure and determined that two additional battery pack lights were required. This modification added the two lights as well as moved two lights from warm to cooler areas.

Summary of Safety Evaluation

This modification added two Teledyne battery pack lights identical to the ones added for 10 CFR Part 50 Appendix R. Two existing battery packs were moved from warm locations to cooler areas with remote heads installed in the original locations. The charging unit, battery cabinet and mounting bracket are all seismically qualified. Power requirements for the new lights were determined to be minimal and were provided from non-safeguards sources. The lights were tested following installation and were added to the plant surveillance program.

62. Modification 90L175, Limit Switch Guideplate Installation

Description of Change

This modification improves the reliability of the remote position indication for various Masoneillan sampling valves. This is accomplished by installing a guideplate under the lower limit switch mounting bracket on each valve. The guideplate prevents the valve positioner from rotating away from the limit switch arms, thus ensuring proper alignment and reliable remote valve position indication.

Summary of Safety Evaluation

Proper installation and post-maintenance testing assures proper guideplate operation. Since the guideplate has no moving parts and no electrical or fluid interface, failure is virtually impossible. If failure were to occur, only the remote indication portion of the respective valve might be affected. The valve itself should remain operable.

63. Modification 90L203 Revision 2, Personnel Protection Grounding

Description of Change

This project is to provide a tested method of grounding de-energized bus bars to ensure personnel safety. This project installed a ball-and-socket grounding clamp, designed and manufactured by AB Chance Company, which will connect the three bus bar phases to station ground. A ball stud will be permanently mounted to each bus bar phase and a portable cable assembly will be utilized to complete the ground. This project is QA-Type III (non-safety related).

The construction standard that was utilized interfaces with existing grounding standards and is consistent with the NSP Grounding Procedures.

No Technical Specification or USAF information is needed for this modification.

Summary of Safety Evaluation

This design change package does not require any modifications to the Technical Specifications, Operations Manual or USAR. Nor does this modification change the functional design or operation of the existing bus system or grounding system. The result of this modification will be increased personnel safety when working on de-energized bus bars.

64. Modification 89Y015, Chlorine System Replacement

Description of Change

This project replaced the existing cooling water treatment system which utilizes chlorine with a new system using sodium hypochlorite and sodium bromide. The two chemicals are combined before injection to the cooling water system to form hypobromous acid which provides an effective biocide control in the piping systems.

Summary of Safety Evaluation

All portions of this modification are non-safety related. Seismic considerations are not required in the physical area of the plant subject to this modification. The injection of the hypobromous acid will not affect the operation of the cooling water system. It will only serve to reduce biological growth. The elimination of the chlorine on site will reduce the possibility of a spill which could affect other safety related systems. This modification only affects piping upstream of the safety class break.

None of the chemicals sodium hypochlorite, sodium bromide, or hypobromous acid are considered as toxic chemicals per NUREG 0570, and 29 CFR 1910.1000 (OSHA) and, therefore, do not affect control room habitability issues addressed in USAR Section 7.8.2.

65. Modification 88L050, Accumulator Discharge Valve Annunciator Bypass

Description of Change

This project is to permanently wire the 11, 12, 21 and 22 accumulator discharge valve position annunciators to show true valve position during all plant conditions. All work associated with this project is QA-Type III, non-safety related and does not change the operation or design of any existing system.

Temporary jumper wires have been installed to remove the SI blocked contact (normally closed) from the alarm circuits. This modification will re-terminate one wire on each alarm circuit to permanently bypass the SI blocked contact.

Technical Specification 3.3.A.1.b requires the reactor coolant system accumulators be operable when reactor coolant system pressure is greater than 1000 psig. SI block occurs when the reactor coolant system pressure falls below 2000 psig. Previously, the windows would alarm only if the SI was unblocked, thus creating a gap from 1000 to 2000 psig where the alarms were inoperable. This modification will remove the SI blocked contact from the alarm circuit so the correct valve position is indicated during all plant conditions.

Summary of Safety Evaluation

Since true valve position will be indicated at all times by the annunciator window, there will be no chance for confusion. Hence, this modification does not increase the consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments.

This modification corrects a previous non-conformance to the Technical Specifications, thus the margin of safety defined in the bases for the Technical Specifications is not reduced.

66. Modification 89L139 Insulation Test Probe Guides for D1 and D2

Description of Change

Tubing was added to the diesel generator engine skids to guide generator bearing insulation test probes.

Summary of Safety Evaluation

The tubing is adequately supported to prevent it from causing harm to the engine.

67. Modification 86L957 Revision 1, Add Pipe Anchor to Support Valve

Description of Change

Piping was anchored closer to safety related component so scope of piping analysis could be reduced.

Summary of Safety Evaluation

The volume control tank gas supply piping analysis was updated.

68. Modification 88L067 Part A1, Safety Injection Pump Performance Curves

Description of Change

Safety Evaluation No. 294 evaluated High Head safety injection pump performance including allowances for previously omitted test instrument inaccuracies. This modification generated new pump performance limit curves to provide adequate margin for future pump degradation and the variability of pump test results. Test instrument inaccuracies will be incorporated into future pump testing.

Summary of Safety Evaluation

The associated accident analysis were reviewed for impact and evaluated. No physical plant changes were required.

69. Revision 16 to the Operational Quality Assurance Plan

Revision 16 to the NSP Operational Quality Assurance Plan was internally reviewed and approved May 18, 1992. We have concluded that this revision does not reduce the commitments of NSP's Operational Quality Assurance Program and does not adversely impact safe operation of the nuclear power plants. Specific changes with the reason for the change and basis for concluding no reduction in commitments [per 10 CFR Part 50 Section 50.54(a)(3)] are presented in Appendix D to the plan. The Operational Quality Assurance Plan, Revision 16, is included in Appendix C of the USAR.

70. Design Basis Document Reconstitution Follow-on Item No. A0307

Description of Change

Change the designation of a distribution panel in the 125 V DC & 120 V AC Instrument Supply on USAR Figure 8.5-1.

Summary of Assessment

The assessment noted that a modification had removed a DC power panel and replaced it with a junction box and changed the fuse rating in the feeder panel to conform with the load and wiring requirements of panel 191. Fuse ratings were verified with the Prairie Island Nuclear Generating Plant "Electrical Coordination Study".

71. Design Basis Document Reconstitution Follow-on Item No. A0104

Description of Change

Change the time delay for initiation of safety injection from five seconds to four seconds.

Summary of Assessment

There are several different values for the time of safety injection flow initiation. The time of initiation of safety injection flow after generation of an "S" signal is four seconds for the large break loss of coolant accident; 25 seconds for the small break loss of coolant accident; and 10 seconds for a main steam line break. These values are being used as inputs for the most recent accident analysis on record. The original design data is compatible with the times used in accident analysis and design documents. This discrepancy has no operability impact.

72. Design Basis Document Reconstitution Follow-on Item No. A0396

Description of Change

Change the operating pressure of the fire protection system from 120 psig to 125 psig per Design Study Number 13.

Summary of Assessment

The USAR uses both 120 psig and 125 psig as the operating pressure in the fire protection system. Design Study Number 13, user 125 psig. The correct pressure for the fire and screenwash pumps is 125 psig.

73. Design Basis Document Reconstitution Follow-on Item No. A0397

Description of Change

Change the minimum pressure at the highest fire protection system hose station from 75 psig to 65 psig per Design Study Number 13.

Summary of Assessment

The USAR uses both 75 psig and 65 psig as the minimum pressure in the fire header, measured at the highest point. The original design goal was 80 psig. Design Study Number 13, uses 65 psig as the minimum pressure and the draft NRC Fire Protection Safety Evaluation comments, noted that the header pressure can drop to 65 psig at the highest hose station in the system, this pressure would only exist with a 2000 gpm demand. The correct minimum pressure at the highest hose station is 65 psig.

74. Design Basis Document Reconstitution Follow-on Item No. A0666

Description of Change

Provide additional information/clarification on the function of the cooling discharge low header pressure switch and its start of the diesel driven cooling water pump in a loss of offsite power scenario.

Summary of Assessment

This function of the cooling discharger header low pressure switch is documented in the AEC Safety Evaluation for Prairie Island and the start of the diesel driven cooling water pumps on loss of station power is discussed in the Prairie Island Technical Specification Basis. The additional information regarding the pressure switch provides clarification on the start signal for the diesel driven cooling water pumps on a loss of offsite power.

75. Design Basis Document Reconstitution Follow-on Item No. A0501

Description of Change

Correct component cooling system USAR inconsistencies; update list of equipment serviced; change nomenclature for consistency; update list of equipment identified as not being isolated, by deleting equipment which is isolated and adding equipment which is not isolated; add the component cooling heat exchangers to the list of components which have safeguards function; update pump descriptions for consistency; and clarify environmental qualification, seismic category, and Quality assurance requirements for component cooling heat exchanger outlet temperature and flow components.

Summary of Assessment

The above discrepancies were identified following review of the appropriate documents, such as Prairie Island Nuclear Generating Plant Operations Manual component cooling description, component cooling flow diagrams, component cooling system design basis document, Regulatory Guide 1.97, Revision 2, 10 CFR Part 50 Section 50.49, and Regulatory Guide 1.89. These discrepancies are typographical/editorial errors, therefore there is no operability impact.

76. Design Basis Document Reconstitution Follow-on Item No. A0532

Description of Change

Enhance guidance, in USAR, for operation during a flood by specifying that transformers 1R, 2R and 1CT1, are to be used to provide offsite power following a design basis flood.

Summary of Assessment

There are more feasible methods available for ensuring offsite power following a design basis flood event such as defeating the appropriate substation breaker control circuitry and providing fault protection at the offsite source, than the installation of temporary jumpers around the respective plant substations. The addition of the more feasible methods for ensuring offsite power following a design basis flood provides enhanced guidance.

77. Design Basis Document Reconstitution Follow-on Item No. A0126

Description of Change

Update Regulatory Guide 1.97 instrument table to reflect the plants regulatory compliance position.

Summary of Assessment

NRC Regulatory Gui·· 1.97 recommends temperature instrumentation be installed at the containment sump; or as an alternative, temperature instrumentation could monitor the RHR heat exchanger inlet, to monitor RHR system operation following an accident. Existing RHR heat exchanger inlet temperature instrument is located outside of the RHR recirculation flow path, therefore existing instrumentation could not be used to monitor RHR system operation. Little benefit would be gained by installing containment sump temperature instrument; and installing RHR heat exchanger inlet temperature instrumentation in the RHR recirculation flow path would be difficult. The NRC approved installation of a temperature detector in the vicinity of the containment sump. However some documents indicate that the RHR heat exchanger inlet temperature instrument is a Regulatory Guide 1.97 instrument but it is not.

78. Design Basis Document Reconstitution Follow-on Item No. A0117

Description of Change

The Refueling Water Storage Tank Volume requirement for accident mitigation/recirculation purposes is 85,000 gallons. This information is added to the USAR.

Summary of Assessment

A review of the design basis reconstitution follow-on-item indicated that no back-up documentation indicating the suitability of the refueling water storage tank volume existed. Subsequently a document index was compiled and sorted for refueling water storage tank calculations. The original calculation was located in the plant records. This follow-on-item is not considered to have an operability or technical specification impact.