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Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401-1927
Telephone (612) 330-5500

June 28, 1990

Submitted pursuant
to 10 CFR 50.71

Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Submittal of Revision No. 9 to the
Updated Safety Analysis Report (USAR)

Pursuant to 10 CFR 50.71(e) we are submitting 13 copies of Revision No. 9 to the Updated Safety Analysis Report (USAR) for the Monticello Nuclear Generating Plant. This revision updates the information in the USAR for the period from January 1, 1989 through December 31, 1989.

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

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

Include in Exhibit B is Revision 14 to the Northern States Power Company Operational Quality Assurance Plan in compliance with 10 CFR 50.54(a). Changes in Revision 14 to the Plan are describe in Exhibit A (Item 50, page 22) of this letter.

Thomas M Parker

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
G Charnoff

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Attachment

Exhibit A

MONTICELLO NUCLEAR GENERATING PLANT

ANNUAL REPORT OF CHANGES, TESTS AND EXPERIMENTS - DECEMBER, 1989

The following sections include a brief description and a summary of the safety evaluation for those changes, tests and experiments which were carried out without prior NRC approval, pursuant to the requirements of 10CFR50.59(b).

1. EE 87-113, Offgas H_2 Analyzer Inlet Traps Drain Line Reroute And Check Valve Installation

Description Of Change:

The A-Train and B-Train offgas recombiner H_2 analyzer sample lines have moisture removal traps that discharge condensate removed from the sample stream back to the main condenser. The slope of the trap discharge line was changed to provide better gravity drainage. Also a check valve and check valve bypass valve was installed on each drain line to prevent reverse steam flow through the traps and into the analyzers during train startup.

Summary of Safety Evaluation:

This modification improves the analyzer reliability, since the most probable failure mode has been moisture intrusion. The check valve could adversely affect analyzer operability if it failed closed, since moisture could backup into the analyzers. This condition would result in low flow alarms and be annunciated in the control room. The bypass valve was installed to address this concern. All components added by this modification are consistent with the existing drain line design.

2. SRI 89-002, Justification for Continued Operation With Current ECCS Room Sump Design

Description of Change:

There is a discrepancy between the as-built condition of the sump pumps and the current USAR design description. This JCO allows the plant to continue operation until the USAR is revised or the sump pump system modified, if required, to accurately reflect the system's design basis.

Summary of Safety Evaluation:

The evaluation of the Residual Heat Removal and Core Spray Systems operability during an internal flooding event concluded that no safety concerns existed and continued plant operation was acceptable.

3. SRI 89-003, Justification for Continued Operation With RHR and Core Spray Minimum Flow Less Than Vendor Recommendations

Description of Change:

NRC Bulletin No. 88-04 addressed a concern regarding inadequate minimum flow protection for safety-related pumps. As part of the investigation into this issue, NSP contacted the safety related pumps vendors, to see if the minimum flow protection was adequate. The vendors confirmed that minimum flow protection was adequate for the HPCI and RCIC pumps, but that the minimum flow protection was not adequate for extended operation of the RHR and Core Spray pumps. During the original plant design the minimum flow lines were sized for 5-10% of design pump flow. This was based on preventing overheating of the pumps. The new vendor criteria for minimum flow lines is 20-30% of design pump flow to prevent pump damage due to hydraulic instability.

Summary of Safety Evaluations:

Continued operation can be justified based on the following:

a. The short amount of time the pumps run with minimum flow is less than 1% of the total time the pump runs.

b. BWR operating experience has not shown any impact on pump life due to minimum flow operation.

c. For small break events that could result in operation on minimum flow, emergency procedures allow the pumps to be secured until injection is necessary.

d. Only certain small breaks would actually require injection from the RHR or Core Spray pumps.

e. Routine surveillance (Section XI IST) and preventive maintenance practices would detect any excessive pump wear.

4. SRI 89-006, Deviation From NFPA Code in EDG Room Sprinkler System

Description of Change:

This SRI provides resolution to PSQA safety concerns related to apparent deviation from NFPA codes in the EDG room sprinkler system.

Summary of Safety Evaluation:

Plant safety is not compromised, since the concerns are resolved through code interpretations valid engineering considerations, and the "defense in depth" fire protection philosophy of the plant.

5. SRI 89-008, Implementation of Revised Pressure-Temperature Curves Based on Reg. Guide 1.99, Rev. 2

Description of Change:

The Monticello reactor vessel pressure temperature curves were updated to Regulatory Guide 1.99, Revision 2. Revision 2 to Regulatory Guide 1.99 (Revision 2) presents a different method for predicting the Reference Temperature of Nil-Ductility Transition (RT_{NDT}) shift due to irradiation, typically increasing the shifts for BWRs.

Summary of Safety Evaluation:

Use of the new pressure temperature curves raises the minimum temperature required for reactor pressurization.

6. SRI 89-013, Isolation of Reactor Water Cleanup Flow to the "B" Feedwater Loop

Description of Change:

RWCU return flow was isolated from the "B" Feedwater Loop. This prevents back flow through a leaking check valve from affecting RCIC and/or HPCI flow.

Summary of Safety Evaluation:

Isolation of RWCU return to one loop of feedwater has a negligible effect on feedwater nozzle fatigue usage.

7. SRI 89-021, Wiring Discrepancy in MO-2063 Open Permissive Logic

Description of Change:

A wiring discrepancy found in High Pressure Coolant Injection (HPCI) Relay Panel 9-39 was evaluated.

Summary of Safety Evaluation:

The wiring discrepancy did not pose a safety concern with respect to the ability of the HPCI system to perform its design function.

8. SRI 89-027, Primary Containment Automatic Isolation Valves

Description of Change:

Update the containment isolation valves table.

Summary of Safety Evaluation:

The Reactor Water Cleanup Return Valve MO-2399 was deleted from the USAR containment isolation table because it does not function as a containment isolation valve. This valve isolates the Reactor Water Cleanup (RWCU) system from the High Pressure Coolant Injection (HPCI) and Reactor Core Injection Cooling (RCIC) systems. Valve MO-2399 closes on the receipt of a Group 3 isolation signal. The associated containment isolation function is performed by feedwater check valves. These valves do not appear on the USAR table since they do not close as the result of an isolation signal.

The Permissible Operating Time for the Combustion Gas Control System (CGCS) valves was revised from 20 seconds to 60 seconds. This time was not consistent with the Maximum Operating Time of 60 seconds given in Monticello Technical Specifications.

9. SRI 89-034, USAR Clarification of Voltage Response Time

Description of Change:

During performance of the ECCS undervoltage functional test it was noted that bus voltage was not restored to "nearly rated voltage" within 2 seconds as stated in the USAR. Current guidance regarding voltage response time was incorporated.

Summary of Safety Evaluation:

The total elapsed time from initiation signal to the last pump automatically starting was within the limits established in the USAR (Table 8.4.1). More current guidance allows 60% of the time between automatic pump starts for voltage to be restored.

10. MOD 82M040, Torus/Drywell Vacuum Breaker Part Replacement Program

Description of Change:

The torus/drywell vacuum breakers were modified to withstand the cyclic loads imposed by the chugging and condensation of a loss of coolant accident.

Summary of Safety Evaluation:

The modification increased the ability of the torus to dryweII vacuum breakers to function and withstand the consequences of LOCA and post-LOCA conditions. The replacement parts for the vacuum breakers meet or exceed the material properties of the existing parts.

11. MOD 85M028, RCIC Drag Valve Installation

Description of Change:

The Reactor Core Isolation Cooling System (RCIC) test return line was modified, by replacing a motor operated globe valve and restricting orifice with a motor operated drag valve. This modification permits cold quick starts of RCIC to simulate an actual discharge to the reactor. The test return line isolation function did not change with this modification.

Summary of Safety Evaluation:

The design integrity of the modified test return line is at least equivalent to the original test return line. RCIC capability to inject full flow to the reactor within 30 seconds has been maintained.

12. MOD 87Z013, Warehouse Facility

Description of Change:

A 33,600 sq. ft. warehouse was constructed along with the required connections/provisions for water, sewer, fire protection and electrical power.

Summary of Safety Evaluation:

The design complies with commercial building codes.

13. MOD 87M042, Roots Blower Removal

Description of Change:

With the completion of Mark I Containment Modifications, the Roots Blowers and Nitrogen Pumpback Containment Isolation Valves are no longer needed to establish a drywell to torus differential pressure. The work performed included removing the roots blowers, the inlet isolation valve CV-7440, nitrogen pumpback containment isolating valves CV-7436 and CV-7437, and associated piping and electrical controls.

Summary of Safety Evaluation:

Pipe caps were installed in place of the isolation valves. The pipe caps are passive components having no failure modes under containment design conditions. Acceptance criteria for the welded caps is zero leakage. Associated testing insured that there is no increase in leakage from primary to secondary containment. Operability testing verified wiring changes were made properly. No potential for future failures exists since no new active components have been added.

14. MOD 87M050, Waterproofing of Control Room Lavatory Pipe Chase Penetrations

Description of Change:

Piping penetrations CS109, CS110, CS111, and CS112 in the floor of the pipe chase in the control room lavatory and the addition of small louvers in the east wall of the control room lavatory were waterproofed.

Summary of Safety Evaluation:

The control room floor is a fire boundary. No new penetrations were made and the installed fire barriers on the cable spreading room side of the floor were not altered or degraded by the waterproofing on the control room side of the floor. The integrity of the fire barriers were not affected or altered by the waterproofing activity.

15. MOD 88Z003, Security Portal Modifications

Description of Change:

A new access control door was installed at the base of the stairway leading from Access Control into the Reactor Building.

Summary of Safety Evaluation:

The new door has the identical fire rating of door #62 and will be installed in such a manner that it will meet or exceed all qualification requirements for door #62. No electrical loads will be modified or added.

16. MOD 88Z013, Replacement of Feedwater Heaters 11A, 11B, 12A and 12B

Description of Change:

Replaced four condenser neck mounted feedwater heaters with new heaters fabricated of materials more resistant to erosion/corrosion. Other erosion/corrosion repairs include sleeving the 7,8,10,12 and 13 extraction nozzles in the low pressure turbines, replacement of the extraction steam lines between the low pressure turbines and the new feedwater heaters, installation of a new door in the turbine building, and relocation of feedwater heaters 12A and 12B level control instrumentation.

Summary of Safety Evaluation:

The new heaters were specified and designed for the present operating feedwater and extraction steam flow conditions. The new heaters are approximately ten feet longer than the original heater. The additional weight for the new heaters required new supports. The new heaters have larger diameter vent lines which are sized to provide non-condensible gas removal equivalent to 1.5% of the extraction steam flow rate. The hangers and supports for the non-condensible vent lines were designed to support the pipes in a flooded condition. The new door is a UL approved fire door and has a three hour rating. The increase in the power requirement for the new level controllers has been incorporated into the load study.

17. <u>MOD 88M019, Replace RHR Pump Start Times</u>

<u>Description of Change:</u>

The existing RHR pump start timers were replaced with time delay relays due to deteriorating performance. The fuses in the modified circuit were replaced.

<u>Summary of Safety Evaluation:</u>

The design was evaluated for electrical and seismic adequacy of the relays, and their effect on the electrical and seismic performance of the panel. Fuse/breaker coordination adequacy of the modification has been shown. The electrical loads added by this change were reviewed for their effect on circuit loading and found acceptable. No operator actions were required in response to the "TIMER ACTIVATED" alarms. The time delay relays were tested after installation to demonstrate acceptable performance.

18. <u>MOD 88M020, Replace MSIV Solenoid Valves</u>

<u>Description of Change:</u>

The existing solenoid manifolds were replaced with a new model furnished by the same manufacturer. The new solenoids are environmently qualified per NUREG 0588. Conduit seals were installed to maintain the qualification of the solenoids.

<u>Summary of Safety Evaluation:</u>

The new components are environmentally qualified per NUREG 0588 and IEEE standards. Qualification was done by General Electric and seismic analysis was done by Nutech to support the seismic qualifications done by General Electric. Bench testing of the solenoid manifolds and operational testing of the Main Steam Isolation Valves was performed to verify operability.

19. MOD 88M023, Annunciator Bypass for 125V Battery

Description of Change:

Installed circuitry capable of bypassing the Battery Charger Supply Undervoltage Annunciator on the 125VDC battery chargers, so that the standby battery charger can be deenergized without activating the alarm. The manufacturer suggested that the battery charger be left in a deenergized state.

Summary of Safety Evaluation:

This modification is considered safety related with respect to the battery charger. Operation of the bypass does not affect safety related operation of the charger. It affects annunciation only. The annunciator circuits and associated changes are not safety related.

20. MOD 88M030, Monorail Over Plug to HPCI Tank Room

Description of Change:

Installed a 600# capacity monorail on 935' Elev. of the Reactor Building. This monorail is used to hoist items from the CRD Pump Room and from the Reactor Building Floor/Equipment Drain Tank Room. The monorail replaces a lifting lug.

Summary of Safety Evaluation:

The monorail has been designed by Bechtel in accordance with "Civil - Structural Design Criteria for the Monticello Nuclear Generating Plant, Rev. 1". The monorail was installed, inspected, and tested using approved plant procedures. Methods, which have been tested and documented, were utilized to protect adjacent safety-related equipment during construction.

21. MOD 88M032, Torus Purge Valve AO 2381 Seal System Air Lines Modifications

Description of Change:

Three instrument tubing lines associated with torus purge valve AO 2381 seal air system were rerouted, supported, and replaced with tubing of a stronger material, thus providing support and protection for the lines and making them less prone to damage.

Summary of Safety Evaluation:

The analyses performed on the new seal air lines verified that code stress allowables were not exceeded and minimum wall thickness requirements were maintained. The new tubing has a higher tensile strength and is harder than the old tubing, thus it is less prone to damage.

22. MOD 88M033, Improved Position Indication for HPCI and RCIC Testable Check Valves

Description of Change:

The position indication device on the HPCI and RCIC testable check valves were replaced with a ferromagnetic sensor, a control base unit and an output module, thus increasing the reliability of the position indication of the testable check valves.

Summary of Safety Evaluation:

No safety concerns exist regarding the impact of the new devices on the AC and DC load studies, the breaker/fuse coordination study, or seismic considerations or the operability of the HPCI and RCIC systems.

23. MOD 88M041, Replace Carbon Steel Nitrogen Piping

Description of Change:

The Primary Containment nitrogen purge vaporizer bypass line is a one inch carbon steel line. During containment inerting, liquid nitrogen comes in contact with this line and could cause failure due to brittle fracture. The carbon steel vaporizer bypass line was replaced with a copper nitrogen gas supply line from the Drywell instrument nitrogen gas supply line header to eliminate a possible failure due to brittle fracture. The new nitrogen gas supply line connects into the old purge vaporizer bypass line valve, DWV-113.

A pressure indicator was installed on the nitrogen supply line for the Drywell instrument header near the nitrogen control panel.

Safety Evaluation Summary:

The possible hazard or safety concern related to this modification involves the potential affects on the Drywell instrument nitrogen supply. With the removal and replacement of the nitrogen vaporizer bypass line there exists a possibility that faulty procedures, faulty installation or a failure in the equipment could cause degraded or loss of Drywell instrument nitrogen supply. Since the Drywell instrument nitrogen supply is backed up by the instrument air system and will auto transfer to air on loss of nitrogen supply, there is no safety concern or possible hazard to the plant.

24. <u>MOD 88M044, RHR Aux Air Compressor Modifications</u>

<u>Description of Change:</u>

The RHR auxiliary air compressor drain line tubing was rerouted and replaced with stronger material to make it less susceptible to damage and over stressing. The drain traps on these lines were replaced with traps more suited for the application. In addition, the safety valves were replaced with valves having a lower set point to protect equipment served by the compressors from being over pressurized.

<u>Safety Evaluation Summary:</u>

The new drain line tubing was qualified for internal pressure, dead weight, thermal and seismic loads. The new tubing has suffient wall thickness and its affect on the galvanic corrosion rate in the air compressor and drain trap is insignificant. The new drain traps are large enough to adequately remove the maximum condensate load of the compressors. The pressure temperature rating of the drain traps envelopes the drain line, design valves. The set points of the new safety valves were checked prior to installation. They were constructed to ASME code requirements.

25. <u>MOD 88M050, Upgrade ATWS Electrical Isolation from Safety Related Buses</u>

<u>Description of Change:</u>

Rewired the inside of the ATWS panels. This rewiring results in all loads in the ARI circuitry being isolated from the safety related buses by coordinated fusing. This improves the electrical fault isolation from the 125 VDC distribution panels.

<u>Safety Evaluation Summary:</u>

This modification rewired an existing fuse in both the Channel A and Channel B of the ATWS ARI valve circuit to improve load isolation from the safety related 125 VDC buses. No new components are added. Only the wiring configuration for the ARI solenoid negative return side was changed. The ARI circuit operation will not change from that which was previously evaluated for safety considerations. This modification has no affect on power distribution, circuit loading, or redundant channel isolation.

26. MOD 88M051, EDG Ventilation Damper Linkage Modification

Description of Change:

The EDG room supply and exhaust air dampers failure mode was changed from fail close to fail open. The recirculation air dampers failure mode was changed from fail open to fail close on loss of instrument air. These changes assure adequate cooling of the EDGs should the instrument air system fail.

Safety Evaluation Summary:

The mod was done to assure electrical equipment located in the EDG rooms would not overheat in the event of a loss of instrument air. Previously this would have resulted in OA dampers closing and room air being continually recirculated. With the dampers in this new configuration, overheating of the room will not occur. However, a concern existed with overcooling the room in winter.

The concern regards the potential of a low EDG room dtemperature due to a loss of instrument air during the winter. With the engine running enough internal heat is generated to maintain the EDG components operable. A failure of the EDG room supply and exhaust air dampers without the engine running would result in the infiltration of cold air. Unit heaters are located in each EDG room. If the unit heaters were not operable, room cooldown would be noted by operations personnel during their normal rounds.

27. MOD 88M063, Replacement of RWCU Check Valves RC-6-1 and RC-6-2

Description of Change:

Replaced RWCU check valves RC-6-1 and RC-6-2 with smaller check valves to assure full disc lift with normal RWCU flow of 83 GPM through each return line.

Safety Evaluation Summary:

The degraded check valves are being replaced with the proper sized check valves for actual RWCU flow conditions.

28. MOD 88M072, Instrument Air Dryer Purge Isolation

Description of Change:

Installed an isolation valve in the purge exhaust of the air dryer which closes automatically if a drop in instrument air pressure is sensed downstream of the dryer. Purge flow was increased.

Safety Evaluation Summary:

The increased purge flow will not affect vital equipment.

29. MOD 89Z003, Part A, Chemistry Sample System Upgrade

Description of Change:

Reactor building sample station C217 and turbine building sample station C213 have been replaced. All new analytical in-line monitors are installed. A new data acquisition system is installed.

Safety Evaluation Summary:

Original design criteria have been adhered to except where the modification improved the design. The racks are seismically II over I anchored. The modification enhances plant performance and material monitoring.

30. MOD 89Z006, Non Class 1E Power Supply Upgrade

Description of Change:

This modification performed the following:

1. Installed a non-divisional 125 KVA UPS system.

2. Installed a 250V battery system.

3. Installed a 1600 KW diesel generator set.
4. Transferred the Emergency Bearing Oil Pump and Emergency Seal Oil Pumps to the 250V battery distribution.

5. Transferred the VAX computer loads to the new UPS system.

6. Reconfigured the computer room air conditioning system so that the computer is adequately cooled during a loss of off site power.

7. Installed HVAC to support the new UPS, 250V battery and the #13 Diesel Generator.

8. Added fire detection in the new UPS and battery rooms, and detection and protection to the new diesel generator enclosure.

Safety Evaluation Summary:

The electrical load study, the electrical protection coordination review, and the updated computer model of the Monticello Plant provide assurance that there has been no degradation of the system by the new installation.

31. MOD 89M011, Post Accident Sample System Upgrade

Description of Change:

Eliminated the gaseous iodine and particulate sampling capability of the PASS.

Safety Evaluation Summary:

The elimination of gaseous iodine and particulate sampling capability was approved by letter dated May 9, 1988 from R J Wright (NRC) to D M Musolf (NSP). Core damage assessment at Monticello is currently performed utilizing Xenon-133 rather than through PASS iodine and particulate sampling. This Xe-133 method has been reviewed and accepted by the NRC.

32. MOD 89M012 Addition of SW Isolation Valves

Description of Change:

Added isolation valves to the Turbine Building Service Water Headers. Drain valves were installed in the Exciter Cooler inlet and outlet service water lines.

Safety Evaluation Summary:

The entire service water system was required to be isolated to perform the modification. The concerns dealt with total isolation of the service water system including loss of fuel pool cooling, loss of RBCCW cooling, loss of computer room cooling, loss of recirculation pumps, and loss of RHR pump seal cooler cooling.

33. MOD 89Z018, Torus Drain Valve Replacement

Description of Change:

The drain valve installed on Torus penetration X-213A was removed and replaced with blind flange.

Safety Evaluation Summary:

Removing the drain valve from Torus penetration X-213A reduces the stresses on the penetration. Installation of the blind flange makes the configuration of penetration X-213A identical to penetration X-213B.

34. MOD 89Z021, 125VDC/ECCS Upgrade

Description of Change:

The electrical distribution, supplying the Low Pressure Cooling Injection (LPCI) system was revised, such that loss of one division of control power, or an internal breaker fault, will not disable both LPCI and Core Spray.

The LPCI Swing Bus was changed from a normally open bus, with half being supplied form division I and half from Division II, to a solidly tied bus supplied from Division I. If degraded electrical conditions exist, the bus will be automatically transferred to Division II. Tie breakers previously used between MCC 133B and MCC 143B were reconfigured as source breakers. The two MCC's were then hard wired together.

Control Power to open the MCC 133B breaker, and initiate transfer, is derived from the Division I 250 volt battery, 125 volt center tap. This provides two independent sources of control power for Load Center 103 breakers and MCC 133B breakers.

A new time delay relay was installed to allow the Division I diesel to re-energize the LPCI bus in the event of normal power failure. If power is not restored within 12 seconds, an automatic transfer to Division II is initiated. This scheme provides for a one way transfer, therefore, restoration back to Division I requires operator action. Alarms were installed to indicated loss of DC control power and also, transfer to Division II source.

Safety Evaluation Summary:

This modification revised the automatic 480 volt transfer scheme of the Low Pressure Cooling Injection (LPCI) system. Two MCC's previously separated during normal operations are now solidly tied together forming one "LPCI Swing Bus".

System functionality and availability are not degraded by solidly connecting the MCC's. The LPCI is a unique system in that both divisions must be available for the system to be considered operable. If the entire LPCI system fails, the large break LOCA can be mitigated by Core Spray. Therefore the fact that the loss of one LPCI MCC may result in the loss of the second LPCI MCC is acceptable. In fact, system availability is enhanced by removing the circuit breakers on the swing bus due by reducing the number of active components. Isolation of Division I and Division II Power is achieved by using two automatic circuit breakers in series to separate the Division I and the Division II load centers from the solidly tied Swing Bus. Internal failure of any one of these breakers or breaker control circuits will not cause the loss of both a load center and the MCC since all breakers are equipped with coordinated automatic over current trip.

The solid tie between MCC 133B and 143B has the capability to be opened for emergency operations or maintenance.

The LPCI swing bus will normally be powered from Division I. A new relay will sense loss of voltage to the swing bus and initiate automatic transfer to Division II after a 12 second delay. This delay will allow the Division I diesel to restore power and avert transfer.

To reduce the probability of propagating electrical faults between divisions, the LPCI Swing Bus transfer scheme, combined with the Core Spray divisions, will meet the applicable portions of IEEE Standard 279, such as the single failure criterion, testability and quality of components.

35. MOD 89M028, HPCI, RWCU, RCIC, FW Temperature Upgrade

Description of Change:

During performance of the 1986 High Energy Line Break reanalysis, it was discovered that the design temperatures for portions of HPCI, RCIC and RWCU lines were not practical. The purpose of this design change was to document the design temperature increases for portions of RWCU, HPCI, RCIC, and feedwater systems. As a result of the temperature increase to the HPCI line, HPCI support TWH-48 was modified in order to maintain code compliance.

Safety Evaluation Summary:

Nutech Engineers has performed stress analyses of the piping and pipe supports affected by the new temperatures. The analyses concluded that all piping and pipe supports are in compliance with the applicable design code for the Monticello Plant except support TWH-48. An operability evaluation of support TWH-48 was performed which determined that the support met operability requirements per IE Bulletin 79-02. However, Nutech recommended that the condition of TWH-48 be improved and TWH-48 was modified to comply with the applicable design codes.

36. MOD 89M045, Feedwater Regulating Valve Stellite Replacement

Description of Change:

Replace Feedwater regulating valve stellite hard faced cage and plug with non-stellited cage and plug.

Safety Evaluation Summary:

The vendor stated that there were non-dimensional changes. The feedwater pump will not be affected.

37. MOD 89Z047, Division I and Division II 250 VDC Ground Detection Panel Upgrade

Description of Change:

The fuse block from the 250 VDC Ground Detection Panel D101 was removed and installed in the 250 VDC Distribution Panel D100. This relocated fuse block provides greater personnel safety during maintenance since full battery fault currents will be no longer available in the alarm panel. A second portion of this modification added test switches in alarm panels D101 and D102, which allow the panels to be tested during plant operation.

Safety Evaluation Summary:

This modification removed the only safety related component from panel D101 allowing the panel to be downgraded to a non-safety panel.

A Bechtel analysis has shown that panel D100 is not seismically degraded by the addition of the fuse block. This modification does not introduce any new failure modes that have not been previously analyzed.

38. MOD 89M058, Condensate Pump Service Water Discharge Globe Valves

Description of Change:

The Condensate Pump Service Water Discharge gate valves were replaced with globe valves of the same size. An abandoned service water line was also cut and capped.

Safety Evaluation Summary:

No items of safety significance were found regarding the globe valves and the removal of the abandoned line.

39. MOD 89M071, Core Reload for Cycle 14 Operation

Description of Change:

Core reload for Cycle 14 operation was installed during 1989. The reactor core was reconfigured by loading 128 fresh fuel bundles, 9 fresh control blades, and 6 fresh LPRMs, and rearranging the remaining components for Cycle 14 operation.

Safety Evaluation Summary:

The possible safety concerns associated with this work are the use of fuel and control blade designs not previously used at Monticello, and the use of a new fuel loading pattern. The fuel and control blades were designed and analyzed using NRC-approved methodology, and have been approved for use by an NRC SER. The fuel loading plan was designed using NRC-approved methodology, and was found to be acceptable if thermal limit Technical Specification changes were approved (License Amendment No. 70) prior to operation in Cycle 14.

40. MOD 89Z084, Heating Boiler Replacement

Description of Change:

The watertube boiler was replaced with a new Johnson firetube boiler including peripheral equipment such as the blowdown tank, feedwater pumps and the stack.

Safety Evaluation Summary:

The heating boiler is built to ASME Code, Section I, 1986 Edition and is stamped by an authorized boiler inspector. The controls comply with NFP 85A-1987, "Standard for Prevention of Furnace Explosions in Fuel Oil and Natural Gas Single Burner Boiler Furnaces", to assure safe operation. The blowdown tank is constructed to ASME Code, Section VIII, 1989. The FRS-Type fuses in the boiler panel don't coordinate with the MCC breaker, however, the miscoordination will only appear under high fault conditions and the upstream coordination was verified to not create any associated circuit problems with systems other than the heating boiler.

41. MOD 89M087, HPCI Venturi Replacement

Description of Change:

This modification replaced the existing HPCI Flow Venturi, FE-3493. The venturi was replaced by a new venturi made of IGSCC resistant material.

Safety Evaluation Summary:

All design standards used in this modification meet the original plant design criteria.

The new venturi is 3/16" smaller in throat diameter than the existing venturi. Because of the difference in venturi size, the setpoints for the existing Barton instrumentation were adjusted. The new venturi was calibrated in a test stand which duplicated the Monticello plant HPCI system piping configuration. The venturi was calibrated from minimum flow to accident flow rate conditions for the Monticello plant. Pre-operational and operational tests verified the operability of the HPCI system as required by Tech. Spec. Table 3.2.1. The response of the Barton instrumentation to the new venturi was monitored during testing, startup and at rated power. No problems were encountered due to the instrumentation adjustments.

Nutech Engineers performed the requisite piping analysis. Seismic Category I design criteria was used and acceptance requirements were in accordance with the Power Piping Code - ANSI B31.1, 1977 Edition through the summer of 1978 Addenda. Two supports were modified as a result of the piping analysis. The venturi condensing pot support and a support on the 1" instrument lines were upgraded to satisfy the design criteria.

42. <u>MOD 89M092, Part A, GEZIP Mechanical Injection Skid Installation</u>

<u>Description of Change:</u>

A prefabricated skid was installed at 931' level of the turbine building. The skid has a 100 gallon capacity tank to contain a zinc oxide and water slurry. The slurry is pumped via one of two positive displacement pumps into the feedpump suction lines at FW-179-1 and FW-179-2 piping connections. The zinc enters the reactor via the feedwater lines and will serve to reduce the amount of CO-60 plate out on the recirculation piping and thus decrease drywell dose rates.

<u>Safety Evaluation Summary:</u>

A load study and a breaker/fuse coordination study was prepared to verify that the load addition of the GEZIP skid to the plant electrical system is acceptable. Skid discharge tubing is designed to IAW ANSI B31.1 and General Electric fabrication specification 23A6325. A floor loading study was prepared to ensure the skid weight is acceptable for its location in the turbine building. An engineering design review was performed to ensure resulting stresses and fatigue usage are negligible at the feedwater piping connection. General Electric has evaluated zinc oxide relative to primary systems compatibility and fuel integrity. They have concluded that no negative effects are associated with the intentional addition of zinc. General Electric has also determined the zinc to be of no impact as an embrittling agent with respect to primary system components. No negative impact will occur as a result of the use of zinc addition and hydrogen water chemistry simultaneously. The impact of zinc addition on radwaste was evaluated and found to be minimal. The safety evaluation concludes that plant safety is not adversely affected by this modification.

43. <u>MOD 89M093, SRV Discharge Line Support MOD</u>

<u>Description of Change:</u>

The Pipe Clamp Support No. NTI-1 on the SRV Discharge Line RV24-10"-HE located in Bay 4 of the Torus was modified to increase the existing design margin. This task involved the installation of a new gusset plate to reinforce the existing pipe clamp. A 2"x4" portion of the original gusset plate was missing. The new gusset plate bridges the portion of the clamp which was missing.

Safety Evaluation Summary:

Nutech Engineers performed the analysis of the SRV Discharge Line Support No. NTI-1. The results showed that the support was operable since requirements of the AISC code were met. Nutech concluded the safety of the support and the SRV discharge line were not affected but recommended that the condition of the support be improved to increase the existing safety margin.

44. MOD 89M095, Replace Check Valve Limit Switches

Description of Change:

Namco limit switches providing disc position indication on the core spray testable check valves, AO 14-13 A and B, were replaced with Honeywell Micro Switches. During closures under only the force of gravity, the Namco switches would not prevent closure of the valve during reverse flow. The Honeywell switches require much less torque than the Namco switches to trip.

Safety Evaluation Summary:

The check valve is a safety related component for pressure retention only. The indication of disc position provides no inputs to safety related functions. Confirmation of Core Spray flow on C03 is provided by Flow Indicators, FI 14-50 A and B. The switches are not safety or Q.A. related. The Honeywell switches have the same electrical rating as the Namco switches (20 Amps at 125VAC). A pre-operational test confirmed proper mechanical operation and proper indication in the control room.

45. MOD 89M096, Improved Hinge Pin Bushings and Counterweight Lever for HPCI-10

Description of Change:

The hinge pin bushings on High Pressure Coolant Injection (HPCI) stop check valve HPCI-10 were replaced. The new bushings are made of stellite for increased resistance to galling. In addition, the counterweight lever for HPCI-10 was permanently removed for personnel safety considerations.

Safety Evaluation Summary:

No safety concerns exist regarding the design, installation, operation and testing of the new bushings with respect to HPCI system operability.

46. MOD 89M097, RWCU, HELB Modification

Description of Change:

Support REWH-201 on line REW3-4"-DB/DBD was modified to eliminate postulated HELB locations from the room behind the RWCU room. In this room are PCAC (Primary Containment Atmosphere Control) valves AO-2386 and AO-2387 as well as the associated air lines for the T-ring inflatable seats of these valves. During a postulated HELB, the service air supply to the PCAC valves may be damaged resulting in leakage from Primary Containment through the inflatable T-ring seats. The resultant Primary Containment leakage could be unacceptable per Technical Specifications.

Safety Evaluation Summary:

Support REWH-201 was originally a Seismic Category II support. The modified support can be classified as a II/I support since the support has been analyzed to seismic category I criteria. According to HELB criteria, break locations for a category II line must be postulated at all fittings, valves, welded attachments to the pipe and piping terminal ends. Modifying support REWH-201 removed break locations outside of the room behind the RWCU room. Consequently, the air supply to the PCAC valves will not be damaged resulting from a break on the RWCU line REW3-4"-DB/DBD. Thus, this modification did not create or result in an unreviewed safety question.

47. MOD 89Z100, LPCI Swing Bus Degraded Power

Description of Change:

New relaying equipment was installed to monitor the LPCI Swing Bus for degraded power conditions while energized from the Division I diesel generator. In the event degraded power conditions are detected a transfer from the normal Division I source to the Division II source will now occur.

New degraded power relays are seismically qualified, solid state, high accuracy, drawout type ITE Relays. These relays are qualified for a mild environment which corresponds to the environmental conditions where the new relay panel is located. Relays are located in a new seismically mounted cabinet on the wall east of MCC 133B. Control power will come from the Division I 250 volt battery, 125 volt center tap. The relays will retain their current state on loss of control power. The new relays have adjustable set points (dropout/pickup), which provides an adjustable dead band. These relays will monitor the incoming power to the LPCI Swing Bus, when supplied by the Division I diesel, and initiate transfer upon a degraded power condition. Bus parameters monitored will be undervoltage, loss of voltage, overvoltage and out of frequency.

Safety Evaluation Summary:

The automatic 480 volt transfer scheme of the Low Pressure Cooling Injection (LPCI) system was modified. New relaying was installed to monitor the LPCI swing bus for degraded power conditions while energized from the Division I diesel generator. In the event, degraded power is detected, an automatic transfer to Division II is initiated.

New high quality, solid state ITE draw out type relays were installed. Control power, from the 250 volt DC center tap which is nominally 135 volts, is within the relay operating range. Relay contacts used in the circuits are within there design ratings. System operating voltages are not expected to exceed relay ratings.

LPCI is not taken credit for in the Appendix R safe shutdown analysis and therefore this modification is not an Appendix R concern. HELB concerns have been analyzed and through special procedure, LPCI valves may be opened allowing decay heat removal through RHR LPCI Mode achieving safe shutdown.

To reduce the probability of propagating electrical faults between divisions, the LPCI Swing Bus transfer scheme, combined with the two Core Spray Divisions, will meet the applicable portions of IEEE Standard 279, such as the single failure criterion, testability and quality of components.

48. MOD 89M101.GEZIP

Description of Change:

Installed a 1-1/2" Line at each Reactor Feed Pump suction pipe and a 1-1/2" line with double isolation valves on each Reactor Feed Pump discharge pipe. The two lines on the suction side of the Reactor Feed Pumps are used as the GEZIP injection taps. All three lines will be used for the "passive" GEZIP design.

Safety Evaluation Summary:

The piping design was reviewed to assure that existing Seismic I piping in the area is unaffected. The three new 1-1/2" lines are HELB lines. A review was also performed to ensure that the new line frequency would not match a high amplitude frequency of the operating Reactor Feed Pump.

49. MOD 89M012, Replace RCIC Steamline Low Pressure Isolation Switches

Description of Change:

Pressure switches PS 13-87A, 13-87B, 13-87C and 13-87D were replaced with switches which have a range of 5-120 psig. The original switches had a range of 50-1000 psig and were used in the RCIC low steamline pressure logic circuit.

Safety Evaluation Summary:

The seismic and environmental qualification reviews concluded that no safety concerns exist with respect to the switches. In addition no safety concerns exist regarding the design, installation and testing of the switches with respect to the operability of the Reactor Core Isolation Cooling (RCIC) system.

50. Revision 14 to the Operational Quality Assurance Plan

Revision 14 to the NSP Operational Quality Assurance Plan was internally reviewed and approved on May 15, 1990. We have concluded that this revision does not reduce the commitments of NSP's Operational Quality Assurance Program and does not adversely impact the safe operation of the nuclear power plants. Specific changes with reason for the change and basis for concluding no reduction in commitments [per 10 CFR 50.54(a)(3)] are presented in Appendix D to the plan. The Operational Quality Assurance Plan, Revision 14, is included in Appendix C to the USAR.