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SUBJECT: Forwards Rev 5 to updated SAR, including description & summary of safety evaluation changes, tests & experiments made under 10CFR50.59 & updated SAR page changes.

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July 28, 1987

Submitted pursuant
to 10 CFR 50.71(e)

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. 5 to the Updated Safety Analysis Report (USAR)

Pursuant to 10 CFR 50.71(e) we are submitting 13 copies of Revision No. 5 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, 1986 through December 31, 1986.

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.

Included in Exhibit B is Revision 11 to the Northern States Power Company Operational Quality Assurance Plan in compliance with 10 CFR 50.54(a). Changes in Revision 11 to the plan are described in Exhibit A (Item 36, page 13) of this letter.

David Musolf
Manager - Nuclear Support Services

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
G Charnoff (w/o Exhibit B)

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Attachments

Exhibit A

MONTICELLO NUCLEAR GENERATING PLANT

ANNUAL REPORT OF CHANGES, TESTS AND EXPERIMENTS - DECEMBER 1986

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 10CRF50.59(b).

1. SRI # 85-011, Removal of Battery Cell(s) from Operable Battery Systems for Analysis and Preventative Maintenance

   Description of Change:
   The change allows for temporary procedural removal of battery cell(s) on operable plant dc systems when abnormalities are detected.

   Summary of Safety Evaluation:
   To maintain battery system operability during battery cell removal, a paralleling battery and accessories are used. The use of paralleling battery contradicts the battery description in the USAR, but will satisfy voltage and capacity consideration established in it. Therefore, overall the battery system operability will not be degraded with the parallel battery in service.

2. SRI # 86-012, Replacement of Standby Gas Treatment Charcoal Filters With Co-Impregnated Charcoal Adsorber Filter

   Description of Change:
   Potassium Iodine (KI) impregnated charcoal adsorbers in the Standby Gas Treatment (SBGT) were replaced with co-impregnated (KI-TEDA) charcoal adsorbers because the new adsorbers are more resistant to aging and poisoning, and therefore may have improved performance.

   Summary of Safety Evaluation:
   The replacement of KI impregnated charcoal adsorbers with KI-TEDA co-impregnated adsorbers will not detrimentally affect the SBGT system. The KI-TEDA impregnated charcoal meets the requirements of Regulatory Guide 1.52, Revision 1 (June 1976) and Revision 2. It also meets the commitments of the Updated Safety Analysis Report (USAR), Section 5.3.4.1.

3. SRI # 86-017, Use of Increased Secondary Stress Allowables on Jet Pump Instrument Line JP-1-1"-DCA

   Description of Change:
   Due to calculated thermal movement during plant heatup and following Safety Relief Valve (SRV) actuations, a potential problem was found to exist with Jet Pump Instrument Line, JP-1-1"-DCA, interfering with SRV discharge line, RV-24A-10. Selected jet pump instrument lines have been re-analyzed using the Mark I Containment Program Augmented Class 2/3 Evaluation. The results of this analysis show that calculated fatigue stresses would meet augmented allowables developed.

Summary of Safety Evaluation:
Section III, Class II or III piping is analyzed using secondary stress allowables based on 7000 cycles of piping motion. Allowables are decreased as the number of cycles increases. The Mark I curve enables the use of smaller numbers of cycles with correspondingly higher values of secondary stress allowables. Using this method the jet pump instrument line will meet secondary stress allowables corresponding to 1500 cycles of piping motion.

4. SRI # 86-018, Locked Valve Discrepancies

Description of Change:
The P&ID's were updated to reflect the proper position for all 30 of the valves appearing as "locked" in the USAR. Determination was made that of the 30 valves indicated locked on the P&ID's, 26 are not required to be locked.

Summary of Safety Evaluation:
Review and comparison of the Monticello P&ID's, plant startup valve checklists and the USAR revealed discrepancies between valves that are indicated to be locked on the P&ID's and those that actually are locked according to the checklists. These discrepancies were resolved.

5. SRI # 86-020, Air Ejector Radiation Monitor Calibration Method

Description of Change:
The USAR description of the air ejector radiation monitor calibration was found during an audit to be out of date. It was revised to describe an improved procedure.

Summary of Safety Evaluation:
Continuous radioactivity monitoring systems are calibrated against appropriate standards and the relationship established between concentration and monitor readings over the full range of the readout device. The method used is consistent with Regulatory Guide 1.21.

6. EE # 86-034, Lifting Eyes for Maintenance of MO2013, LPCI Throttle Valve

Description of Design Change:
Two lifting eyes were installed in the West Shutdown Cooling Room over valve MO2013 to allow maintenance of the valve to be performed.

Summary of Safety Evaluation:
There are two safety concerns associated with installation of the padeyes; heavy loads and loading on the ceiling.

The only equipment located in the West Shutdown Cooling Room is associated with the B Loop of RHR. This loop would be inoperable if MO2013 was disassembled.

A load drop would not disable the RHR Loop A. Load on the ceiling would not change since the valve is supported from the ceiling.

7. EE # 86-037, High Energy Line Break in the SJAE Area Proposed Fixes

Description of Design Change:
The hatch plug located at the end of the railway bay at the end of the 931' elevation of the Turbine Building was removed to create a vent path in case of a high energy line break within the Steam Jet Air Ejector (SJAE) room. The hatch plug was replaced with reinforced steel plate.

Summary of Safety Evaluation:
Removal of the hatch plug may increase the radiation levels in the hatch plug area. The necessary postings were initiated.

8. DC 77-063, Addendum 1, Remove Reactor Feedwater Nozzle Cladding and Install Improved Feedwater Spargers

Description of Design Change:
During removal of the feedwater nozzle cladding, an inspection of the 45^o feedwater nozzle after completing the safe-end and deep bore cuts revealed that the surface finish was rougher than required by the machining drawing. To correct the situation, it was necessary to take a 0.060-inch diametral skin cut in the affected areas.

Summary of Safety Evaluation:
The stress report indicated that surplus reinforcement area in the nozzle after machining to original dimensions was 26.38 in^2. The skim cuts removed less than 0.3 in^2 of the surplus area. The stress report also indicated that the minimum required average wall thickness in the safe-end was 0.543 inches. Wall thickness after completing skim cuts was 0.671 inches, The skim cuts, therefore left the nozzle and safe-end in an acceptable condition.

9. DC 81-021, Part 4 Addendum 4, CRD Scram Discharge System Modifications

Description of Design Change:
Seismic Category I Review Program Modifications to the Control Rod Drive (CRD) Scram Discharge system.

Summary of Safety Evaluation:
The modifications provide an increased margin of safety in the event of a seismic occurance.

10. DC 82-023, Addendum 1, Seismic Category I Piping Review Program - RHRSW System

Description of Design Change:
Seismic Category I Piping Review Program Modifications to the Residual Heat Removal Service Water (RHRSW) system.

Summary of Safety Evaluation:
The modifications provide an increased margin of safety in the event of a seismic occurance.

11. DC 82-055, 82-055 Addendums 1 and 2, Seismic Category I Piping Review Program - ESW System

    Description of Design Change:
    Seismic Category I Piping Review Program Modification to the Emergency Service Water (ESW) system.

    Summary of Safety Evaluation:
    The modifications provide an increased margin of safety in the event of a seismic occurance.

12. DC 83-014, 83-014 Addendums 1, 3 and 4, Seismic Category I Piping Review Program - RHR System

    Description of Design Change:
    Seismic Category I Piping Review Program Modification to the Residual Heat Removal (RHR) system.

    Summary of Safety Evaluation:
    The modifications provide an increased margin of safety in the event of a seismic occurance.

13. DC 83-023, 83-023 Addendums 1 and 2, Seismic Category I Piping Review Program - HPCI System

    Description of Design Change:
    Seismic Category I Piping Review Program Modifications to the High Pressure Coolant Injection (HPCI) system.

    Summary of Safety Evaluation:
    The modifications provide an increased margin of safety in the event of a seismic occurance.

14. DC 83-024, Seismic Category I Piping Review Program - Primary Steam System

    Description of Design Change:
    Seismic Category I Piping Review Program Modification to the Primary Steam system.

    Summary of Safety Evaluation:
    The modifications provide an increased margin of safety in the event of a seismic occurance.

15. DC 83-025 Addendum 1, Seismic Category I Piping Review Program - SBLC System

    Description of Design Change:
    Seismic Category I Piping Review Program Modification to the Standby Liquid Control (SBLC) system.

    Summary of Safety Evaluation:
    The modifications provide an increased margin of safety in the event of a seismic occurance.

16. DC 83-026, 83-026 Addendum 1, Seismic Category I Piping Review Program Core Spray System

Description of Design Change:
Seismic Category I Piping Review Program Modifications to the Core Spray system.

Summary of Safety Evaluation:
The modifications provide an increased margin of safety in the event of a seismic occurance.

17. MOD 83-069, Etched Disc Floor Drain Filter System

Description of Design Change:
The floor drain filter-demineralizer vessel was replaced with a filter-demineralizer vessel that includes etched disc filter elements.

Summary of Safety Evaluation:
The new system is designed in accordance with Regulatory Guide 1.143.

18. MOD 83-112, Appendix J Modifications

Description of Design Change:
In order to bring Monticello into compliance with the type C testing requirements of 10 CFR 50, Appendix J, various piping systems in the plant were modified. Modifications included addition of leak test connections, block valves and manual containment isolation valves.

Summary of Safety Evaluation:
Existing system functions were not affected by addition of the new valves. All new valves are manually operated and controlled by normal administrative controls. The new valves meet the original design requirements of the systems in which they were installed.

19. MOD 84-074 Addendum 1, Replace XDV-4, Reactor Vessel Bottom Head Drain Valve

Description of Design Change:
The reactor pressure vessel bottom head drain line was determined to be overstressed. This modification added two new anchors and modified existing supports to bring the line into compliance with all applicable code criteria.

Summary of Safety Evaluation:
The modified and new pipe supports will provide an increase margin in safety in the event of a seismic occurance.

20. MOD 84-078 Part A, Part A Addendum 1 and Part A Addendum 1 Revision 1, Hydrogen Water Chemistry

Description of Design Change:
This modification includes 1) adding a sample line dryer to the effluent gas stream to each of the six offgas hydrogen analyzer sample lines, 2) adding connecting taps for future use on the offgas piping downstream of the 24-inch, 2-minute delay, pipe and to the suction of

the condensate pumps, and 3) modifying the primary containment isolation logic so that the reactor water sample valves isolate on Group 3 signals as well as Group 1 signals.

Summary of Safety Evaluation:
The sample dryers installed in the hydrogen analzer effluent stream are designed to increase the reliability of the analyzers by removing moisture from the sample stream line.

The connection to the offgas system and to each condensate pump suction line is intended for future use, with the addition of the permanent hydrogen water chemistry system.

The primary containment isolation logic modification will preclude the possibility of losing reactor water inventory via the path from the reactor water sample line to the hydrogen water chemistry verification station and back to the reactor water cleanup (RWCU) system during events when the RWCU system is isolated by a Group 3 signal.

21. MOD 84-079 Addendum 1, Installation of Lifting Device for B RHR Room Heat Exchanger Shield Plug

Description of Design Change:
A heavy loads monorail lifting device was installed to be used for removal of the concrete shield plug in the reactor building floor above the B RHR heat exchanger. The addendum revised the original design with respect to the design load of the monorail.

Summary of Safety Evaluation:
The monorail was designed to transport a 10,000 lb. load. Re-analysis has shown that it is acceptable to increase of the monorail load design from 10,000 lbs. to 12,000 lbs.

22. MOD 85-013 Part B, Reactor Vessel Level Instrumentation

Description of Design Change:
This modification replaced the condensing chamber/temperature equalizing reference columns inside containment with "cold" reference columns outside containment to eliminate the effect of high drywell temperature on the reactor vessel level indication. The level indicating type differential pressure switches were replaced with differential pressure transmitters which send signals to six new trip cabinets. These trip cabinents intertie with the existing plant logic systems so that the functions remain the same.

Summary of Safety Evaluation:
The original code of construction for piping installation was the USA Standard Code for Pressure Piping, "Power Piping", USA B31.1.0-1967 Edition. The piping modification meets the requirements of ASME Code, Section III for Class 1 and 2 piping. ASME Code, Section XI allows use of this later code edition since it meets the requirements of the original code of construction.

The level transmitters are qualified to IEEE 344-1975 for seismic conditions and IEEE 323-1974 for environmental conditions. The trip cabinets are seismically qualified to IEEE 344-1975 and are seismically anchored. The cable is qualified to IEEE 323-1974 and IEEE 383-1974 for the worst case environmental conditions at Monticello.

The instrument piping is located and seperated to prevent failure of one line from inducing failure in the other line.

The improved system results in a reduction in the error in vessel level indication during a loss of coolant accident.

23. MOD 85-014, New Startup Transformer, 2R

Description of Design Change:
This project replaced the existing #11 Auxilary Transformer with a new 34.5-4.16 kV transformer, designated 2R, with an automatic load tap changer. A new 345-34.5 kV transformer designated 2RS, supplies power to transformer 2R from the #1 345-34.5 kV bus in the substation switchyard. Both transformers have low internal impedance to help reduce voltage drop on the 4.16 kV system during large motor starts.

A new transformer fire sprinkler piping network was installed to accommodate the increased transformer size.

Summary of Safety Evaluation:
Having the 4.16 kV system supplied by the substation instead of being directly tied to the generator will allow the generator to be operated through its full range of rated voltage without exceeding 4.16 kV system voltage operating limits. This modification also provides an additional immediate access source of offsite power to the plant safeguards buses.

24. MOD 85-016 Part B, Appendix R - Alternate Shutdown System - Panels and Cabling

Description of Design Change:
An Alternate Shutdown System (ASDS) was installed and certain Division II cables were rerouted to assure safe shutdown in the event of a fire in either the control room or cable spreading room as required under Appendix R to 10 CFR Part 50.

Summary of Safety Evaluation:
The panels installed in the EFT building and turbine building are qualified to IEEE 344-1975 for seismic accelerations. The cable used in this modification was qualified to IEEE 323-1974 and IEEE 344-1974 for the worst case environmental conditions at Monticello. All new ASDS components that interface with existing systems were designed to the same design criteria as existing safety systems.

The ASDS has no impact on the probability or consequences of other accidents evaluated for the Monticello Nuclear Generating Plant. Isolation and electrical separation of the ASDS panel from the control room, except when in use in the event of a fire in the control room or cable spreading room, is ensured by the panel design. The ASDS design was reviewed and approved by the NRC Staff in a letter from D B Vassallo (NRC) to D M Musolf (NSP) dated September 11, 1985. This modification represents a significant improvement in the ability of the plant to respond to a fire in the control room or cable spreading room.

25. MOD 85-019, Class 1E 120V Instrument AC System

Description of Design Change:
This project provided the plant with a new Class 1E 120V Instrument AC Uniterruptible Power System. This project installed (2) Class 1E inverters to provide Division I and Division II 120V AC Uniterruptible Power Source. Each inverter feeds a Class 1E Distribution Panel containing 24 circuit through a 200 amp Class 1E fused disconnect switch. All Class 1E loads as identified by the plant have been moved from instrument AC Panels Y10, Y20, Y30 and B44P to the new divisional instrument Panels Y70 and Y80. In addition to its Class 1E distribution panel, the Divisional I inverter will temporarily power instrument AC Panel Y10. Likewise, the Divisional II inverter will temporarily power instrument AC Panel Y30. This temporary condition will exist until a new Non-1E Instrument AC System can be designed and installed. Current plans call for the Non-1E system to be completed during the 1987 outage.

Summary of Safety Evaluation:
The Class 1E 120V Instrument AC System was installed to the requirements of Regulatory Guide 1.100 and IEEE 344-1975. The Class 1E instrument power system was designed to meet the requirements of 10 CFR 50 appendix R and Regulatory Guide 1.75 seperation criteria. All Class 1E components have been procured to IEEE 344-1975 Seismic Qualification of Safety Related Equipment. The system was designed that a single failure of one of these components will not disable the output of the system.

26. MOD 85-021, HVAC System - EFT 3rd Floor

Description of Design Change:
The 3rd floor of the EFT building is cooled by a non-safety related air conditioning unit. The purpose of the unit is to provide cooling for the ASDS panels and other equipment in the room. A fresh air supply fan was also installed for human occupancy.

Summary of Safety Evaluation:
The 3rd floor of the EFT building is not part of the EFT system. The fresh air fan does not have an isolation damper. In the event of a fire in the control room and a high radiation or toxic gas release, the operators would wear self-contained breathing apparatus if it was necessary to work on the third floor.

27. MOD 85-031, Feedwater Control System Replacement

Description of Design Change:
The existing electronic analog feedwater control system was replaced with a fault tolerant digital feedwater control system. The new system provides the same single and three element reactor level control functions for the main and startup control valves as the existing system. Additional reliability is provided by redundant controllers and sensor (transmitter) failure protection.

As part of this project, the power source for the reactor pressure transmitters was changed to the Division I and II instrument power supplies and the transmitters and control room indicators were recalibrated for a range of 0-1500 psig. (This satisifies a NUREG 0737/RG 1.97 commitment.) Also, the control room indication was made independent of the feedwater control system so that reactor pressure information will be available regardless of the status of the digital feedwater control system.

Summary of Safety Evaluation:
In the USAR, two failures of the feedwater control system are considered: failure to maximum flow and failure to zero flow. Failures of the new system in these modes are no different than failure of the original system. Should the new system fail in an as-is manner, a gradual increase or decrease in reactor level would occur which would ultimately result in a reactor scram or turbine trip; both of these events have been previously analysed. Therfore, the USAR analysis is valid for the new system.

The interlocks to the recirc speed control system and Rod Worth Minimizer have been designed to provide the same failure modes as the existing system. Operating procedures recognize the possibility of these events and provide corrective action.

28. MOD 85-041, Core Spray System Modification

Description of Design Change:
The core spray piping, safe ends, and testable check valves were replaced with material less prone to intergranular stress corrosion cracking.

Summary of Safety Evaluation:
The core spray piping and safe end were designed and installed in accordance with ASME Boiler and Pressure Vessel Code and ANSI B31.1 Code for Pressure Piping and Power Piping.

The results of a stress analysis performed on the core spray safe end and nozzles concluded that the safe end and nozzle meet all ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition, stress intensity limits to the design and hypothesized conditions. An ASME Class 1 fatigue anaylsis and an ANSI B31.1 analysis was performed to ensure conformance with the code and original design requirements.

The core spray system performance did not change because the same size piping and configuration was used. There was no change in operating parameters as a result of this modification.

29. MOD 86-003, Increase SRV Setpoints

Description of Design Change:
The setpoints of the main steam line safety/relief valves (SRVs) were raised 12 psi to increase the difference between normal operating pressure and the SRVs opening setpoint pressure. This increase in differential pressure, known as simmer margin, reduces the possibility of SRV pilot valve leakage and the potential for unplanned plant

shutdowns resulting from excessively leaking SRVs. The simmer margin increase also allowed a 12 psi increase in the opening setpoints of the low-low set SRVs thereby reducing challenges to this safety system.

Summary of Safety Evaluation:
General Electric Company analyzed the effects of raising the SRV setpoints and low-low set SRV setpoints and concluded that these changes have no impact on any accident analyses or reactor safety system. The results of the GE analysis are documented in NEDO-30771. Among the items that this analysis includes are 1) for the worst overpressure transient peak reactor pressure remains below the ASME Code limit of 1375 psig, 2) SRV discharge line and torus shell loading stress analyses were performed using an 1150 psig SRV setpoint therefore all stressses remain below code allowables with the new 1120 psig SRV setpoint, and 3) containment pressure and temperature responses to LOCAs remain unchanged with the higher SRV setpoint.

30. MOD 86-004, Automatic Depressurization System Logic Modifications

Description of Design Change:
NUREG-0737, Item II.K.3.18 required modification of the automatic depressurization system (ADS) initiation logic to eliminate the need for manual reactor depressurization for certain potential accidents such as line breaks outside of containment. In order to meet this requirement, the high drywell pressure permissive was removed from the ADS initiation logic. Modification of the RHR and Core Spray pump start logic was also required. Bypass timers were added to bypass the low reactor pressure pump start permissive if it had not been satisified after a 20 minute period of low-low reactor vessel level.

Summary of Safety Evaluation:
The requirement to eliminate the need for operator action to manually depressurize the reactor in the event of a line break outside continment is met by this modification. Removal of the ADS high drywell pressure permissive has expanded the capabilities of this system to cover such events. This has improved overall safety sytem automatic response to accidents and ensures adequate core cooling over a wider range of line breaks. Modification of the RHR and Core Spray pump start logic by the bypass timer addition ensures timely pump start and therefore ADS initiation. GE has determined the bypass timer setting per report AE-79-0884. These logic modifications meet the single failure criteria of IEEE-279.

31. MOD 86-006, SRV Air Supply Upgrade

Description of Design Change:
This portion of the safety/relief valve (SRV) air supply upgrade project involved installation of two new Seismic Category I air supply lines inside the drywell from primary containment penetration X-34A and X-105B(F) to SRVs F and B respectively. Manual isolation valves and caps on the lines were installed on the outside of the containment penetrations to facilitate future leak rate testing.

Summary of Safety Evaluation:
The new air lines were designed and installed in accordance with the power piping code ANSI B31.1-1977. Leak testing of the air lines and valves after installation revealed zero leakage. After leak testing the lines were also sealed inside containment pending continuation and completion of the project.

32. MOD 86-011, Instrument Air Compressor Replacement

Description of Design Change:
This modification involved the replacement of the existing #11 Instrument Air Compressor and the modification of the instrument air control scheme.

Summary of Safety Evaluation:
The plant electrical load study has been reviewed for adding the service to the new #11 air compressor from essential MCC B34. The new load will not degrade the voltage below acceptable limits. The existing power feed for compressor #11 was dropped from essential MCC B33 and was abandoned, which will increase the available capacity of this load center.

The new air compressor will increase system capacity and reliability. The new control scheme will simplify operation and add to the versatility of the system.

33. MOD 86-025, Core Reload for Cycle 12

Description of Design Change:
This modification included changing the fuel loading pattern to that required for Cycle 12, installation of a new fuel type, and a change in the computer program databanks used for core monitoring and core prediction.

Summary of Safety Evaluation:
The possible safety concerns associated with this core reload and computer bank installation are 1) differences between new fuel and fuels used previously, 2) reactor cycle specific transient analysis, 3) reactor cycle specific accident and LOCA analysis, and 4) improper installation and verification of computer databanks.

The new fuel type is essentially the same as the old fuel type, except for the addition of a barrier. The specifics of bundle design and nuclear characteristics are discussed in GE Fuel Bundle Design Report EDB No. 1546 dated July 1985. The barrier bundles are exempt from PCIOMR guidelines in accordance with GE documents NEDS-10456-PC, Revision 4, March 1985.

The Cycle 12 transient analysis, contained in GE report 23A4754, February 1986, entitled "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Plant Reload 11 (Cycle 12), concludes that all pertinent transient design criteria are met and that the specified operating flexibility options are available.

The GE accident and LOCA analyes, NEDC-31158P and 23A4754, that the new core design meets LOCA and other pertinent accident design criteria.

The process computer software has no direct effect on the safety of the plant. To ensure that misleading information is not presented that could lead to inadvertant operation outside of the limits imposed by Technical Specifications, an engineer manually spot checked several key data arrays in the new and the old databanks to verify implementation of the proper data exchanges.

34. MOD 86-032, HELB Piping Modification for Emergency Service Water

Description of Design Change:
To eliminate the possibility of a loss of Emergency Service Water (ESW) supply to the diesel generators due to a high energy line break (HELB) in the condensor area, manual isloation valves were installed in the intake structure on ESW lines. These valves provide a means to isolate the break and provide cooling to the diesel generators from ESW pumps. The main steam pipe chase is another HELB area that would have affected the ESW system B loop due to a break. To correct this, an existing valve was relocated from the steam chase to the torus area. With the valve in the new location, cooling of Division II core spray and RHR pumps and room cooler is still possible with service water by using the relocated isolation valve.

Summary of Safety Evaluation:
Two new normally closed valves were installed and one valve was relocated on the ESW lines. These modifications were done for the following reasons, 1) a high energy line break in the condenser area or main steam pipe chase could affect emergency service water flow to the diesel generators for emergency cooling, and 2) service water flow to emergency core cooling system pump motors and room coolers could be interrupted by a high energy line break in the main steam pipe chase. These modifications result in an improved ability of the plant to cope with high energy line break events.

35. MOD 86-035, Replacement of Gaseous Chlorination System

Description of Design Change:
The original gaseous chlorine system was removed and replaced with a liquid sodium hypochlorite system.

Summary of Safety Evaluation:

Gaseous chlorine can be lethal in concentrated doses. Sodium hypochlorite is not. The hypochlorite system chemicals are not combustible.

Since sodium hypochlorite is a corrosive liquid, sodium hypochlorite system piping is PVDF (polyvinylidene fluoride) to within four pipe diameters downstream of the injection point.

36. <u>Change to Operational Quality Assurance Plan Appendix C</u>
Revision 11 to the NSP Operation Quality Assurance Plan was internally reviewed and approved on June 3, 1987. We have concluded that this revision does not reduce the commitments of NSP's Operation 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 11, is included in Appendix C to the USAR.