NSP

Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

October 23, 1998

10CFR 50.71(e)
10CFR 50.59(b)(2)

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

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

Submittal of Revision No. 16 to the Updated Safety Analysis Report

Pursuant to 10CFR Part 50, Section 50.71(e), a signed original and 10 copies of Revision No. 16 to the Updated Safety Analysis Report (USAR) for the Monticello Nuclear Generating Plant are hereby submitted. This revision updates the information in the USAR for the period up to August 1, 1998.

A substantial number of the changes in this general update were identified during the initial activities associated with a special USAR Review Project described in NSP's September 26, 1997 letter to the NRC. These changes reflect: 1). An effort to include in the list of references at the end of each USAR section all documents cited in the text, 2). Editing the list of references to accurately reflect the title, author, and date as listed on the reference itself, and 3). Correction of various word processing errors (e.g. misspellings, incorrect punctuation or capitalization, omissions of characters, sentence structure, etc.). It should be noted that the initial phase of the USAR Review Project also includes an effort to clarify references to codes and standards. Because this effort was not completed in time to be fully reflected in this revision, additional changes to the reference descriptions at the end of each section will be made at the next general USAR update or earlier. Because 3-hole punching has caused obliteration of some sidebar information in previous revisions, the margins for all sections were moved to eliminate this problem. Because of the pagination changes that resulted, most USAR sections have been reprinted and are included in this transmittal. Exhibit C contains the USAR page changes for Revision 16 and instructions for entering the pages.

Exhibit A, "Monticello Nuclear Generating Plant Report of Changes, Tests and Experiments" is the periodic report of changes, tests and experiments required by 10CFR Part 50, Section 50.59(b)(2).

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Exhibit B, "Report of Changes to Licensee Docketed Commitments", provides a brief description and summary of changes to NRC commitments identified to be reported to the Commission in accordance with guidance provided in the Nuclear Energy Institute (NEI) document titled "Guideline for the Managing of NRC Commitments", Rev. 2, December 19, 1995.

This letter contains no new NRC commitments.

Please contact Sam Shirey. Sr. Licensing Engineer at (612) 295-1449 if you require additional information related to this submittal.

M F Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

cc: Regional Administrator – III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
J. Silberg (w/o Exhibit C)

Enclosures: Affidavit to the US Nuclear Regulatory Commission
Exhibit A - Monticello Nuclear Generating Plant Report of Changes, Tests and Experiments
Exhibit B - Monticello Nuclear Generating Plant Report of Changes to Licensee Docketed Commitments
Exhibit C - USAR Revision 16 Changes

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT DOCKET NO. 50-263

Submittal of Revision No. 16 to the Updated Safety Analysis Report

Northern States Power Company, a Minnesota corporation, by letter dated October 23, 1998 hereby submits Revision Number 16 to the Monticello Updated Safety Analysis Report. This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By M F Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

On this 23rd day of October, 1998 before me a notary public in and for said County, personally appeared Michael F. Hammer, Plant Manager, Monticello Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, and that to the best of his knowledge, information, and belief the statements made in it are true.

Marcus H Voth

Marcus H. Voth
Notary Public - Minnesota
Wright County

MARCUS H. VOTH
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2000

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Document Type	Document Number	Revision	Title
NOTE: The following issues are new or revised			
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9703	USAR-01.02	16	INTRODUCTION AND SUMMARY PRINCIPAL DESIGN CRITERIA
9703	USAR-01.03	16	INTRODUCTION AND SUMMARY SUMMARY DESIGN DESCRIPTION AND SAFETY ANALYSIS
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9703	USAR-D.4	16	PRE-OPERATIONAL AND STARTUP TESTS SUMMARY OF PRE-OPERATIONAL TEST CONTENT
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EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

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

1 Justification for Continued Operation With Current ECCS Room Sump Pump Design - Final Evaluation (SRI 89-002, Addendum 1)

DESCRIPTION:

This Safety Review Item addendum addresses additional word changes to USAR Section 10.3.6.3.2, which correct an error made when USAR Revision 13 changes were made to resolve discrepancies in the Reactor Building Radwaste Drainage System description. The wording did not clearly state that the power supply for each of the 100 gpm ECCS room sump pumps comes from the same essential switchgear division as the equipment in that room.

SUMMARY OF SAFETY EVALUATION:

The design basis reconstitution program documentation on internal flooding verifies that the installed systems meet the actual design criteria. Therefore, there is no safety significance to changing the USAR to reflect the as-built configuration of the corner room and reactor building floor drain sumps.

2 Hydrogen Water Chemistry System Oxygen Injection Test to Determine Oxygen Concentration at the Inlet to the Off-Gas Compressors (SRI 93-019)

DESCRIPTION:

This Safety Review Item addresses a special test (Test Procedure No. 8889 "Test to Correlate Off-Gas Compressor Inlet Oxygen Concentration to Recombiner Outlet Oxygen Concentration") to empirically determine the dissipation characteristics of brief high-Oxygen transients downstream of the Off- combiners.

SUMMARY OF SAFETY EVALUATION:

The test procedure and temporary test apparatus are not described in the USAR. No Technical Specification requirements or NRC regulations are applicable to the Oxygen injection test. This activity did not increase the probability or the consequences of an accident or malfunction previously evaluated. Therefore, performance of this test does not constitute an unreviewed safety question.

3 Core Spray Pump Motor Test Without Water Cooling (SRI 94-003, Rev 0, Addendum 1)

DESCRIPTION:

This Safety Review Item addendum documents an evaluation of Test 8898, which was performed to obtain data for an evaluation of minimum cooling water requirements for the Core Spray pump motor thrust bearings (SRI 95-002, Rev 1 and CA-96-106, Rev 1).

SUMMARY OF SAFETY EVALUATION:

The test procedure and temporary test apparatus are not described in the USAR. Test 8898 involved operation of the 12 Core Spray pumps in the same manner that it is operated for quarterly test 0255-03-III except that thermocouples were installed inside the thrust bearing oil reservoir, and cooling water flow was secured. Limits were established for maximum metal and oil temperatures and vibration to provide protection of the pump motor during the test. This activity did not increase the probability or the consequences of an accident or malfunction previously evaluated. Therefore, performance of this test does not constitute an unreviewed safety question.

4 SBGT Flow Control and Off-Gas Dilution Flow Control Changes to Manual Mode of Operation (SRI 94-008 Supplement)

DESCRIPTION:

This Safety Review Item supplement addresses an evaluation performed to support the changing of the normal mode of operation of the SGTS and the Off-Gas Dilution Air System flow control from automatic to manual with the control valves full open. These changes are intended to reduce susceptibility to component failure and improve Secondary Containment capability during accident conditions.

SUMMARY OF SAFETY EVALUATION:

The operation of these systems in manual mode does not affect the design/operational basis addressed by the USAR. This method of flow control is acceptable because it places the system in conditions of operation that would exist during a DBA and assures the maximum available flow for maintaining Secondary Containment integrity under all assumed accident conditions. Wording enhancements are made in the Performance Analysis discussion of USAR Section 5.3.5 to reflect the adequacy of normal operation with full open flow control valves.

5 Pipe Support Drawing Revisions to Reflect As-built Conditions (SRI 96-010, Rev. 0)

DESCRIPTION:

This Safety Review Item evaluates corrective actions made to assure that as-built pipe supports and associated drawings were in agreement. These corrective actions were the result of ISI visual examinations. Non Conformance Report 95-141 concluded that in certain cases the drawing was incorrect, unclear, or the as-built condition was acceptable. In these cases, the recommendation was to leave the support as found and revise the drawing accordingly. The other drawing discrepancies were addressed by changing the support to match the drawings.

SUMMARY OF SAFETY EVALUATION:

This SRI shows that these drawing revisions do not negatively affect compliance with plant licensing requirements. Modifications did not change the design or licensing basis of the plant.

6 1996 Follow-On-Item (FOI) Identified USAR Changes (SRI 96-016, Rev. 1, Addendum 4)

DESCRIPTION:

This revision of addendum 4 to the Safety Review Item provides additional information and clarification to the USAR which are related to FOI that result from the Design Basis Documentation Program. This addendum addresses the upgrade of Core Spray and RHR piping at the primary containment penetration as allowed by criteria established in NUREG-0800.

SUMMARY OF SAFETY EVALUATION:

There are no new physical or design changes as a result of this evaluation addendum revision. The design of the portions of the RHR and Core Spray system between the containment penetrations and the first isolation valves meet the requirements of MEB 3-1 of SRP 3.6.2. Therefore, a high energy line break (HELB) need not be postulated in those piping portions.

7 Safety Evaluation for Revision to Off-Site and Control Room Dose Analysis Methodology for Analysis of MSIV Leakage for Power Rerate (SRI 96-021)

DESCRIPTION:

This Safety Review Item addresses USAR wording changes to reflect modifications made to the main condenser and drain piping to improve the collection of MSIV leakage. Minor modifications to supports are involved (see Mod 96Q015, Part A & B).

SUMMARY OF SAFETY EVALUATION:

Methods consistent with the Seismic Qualification Utility Group (SQUG) seismic equipment qualification guidelines have been used to assure that the MSIV leakage collection path would remain intact following a design basis earthquake. This evaluation of the main condenser and steam line drain piping was performed to assure that credit for its integrity can be taken when the NRC approves NSP's request to use a new GE methodology (NEDC-31858, Rev 2) for calculating post-LOCA off-site and control room doses attributable to MSIV leakage. The new GE methodology would be applied as part of the plant power rerate program.

8 EOP Guidance For Restoration of Safeguards RPV Water Level Indication After Flashing Conditions in the Drywell. (SRI 96-040)

DESCRIPTION:

The purpose of this Safety Review Item is to enhance current guidance regarding the availability of safeguards reactor water level indication after possible flashing conditions. The current EOP guidance is that RPV water level indication should be considered "unknown" until the reference columns are manually back-filled to confirm they are full. The new guidance takes advantage of modifications (See Modification 83Z117) that minimize the drywell temperature effects on RPV water level indications.

SUMMARY OF SAFETY EVALUATION:

The suggested guidance takes advantage of modification 83Z117 "Reactor Vessel Water Level Instrumentation," which rerouted the reference columns such that their vertical drop in the drywell is approximately 9-inches. This significantly reduces the potential effects of flashing of these columns and results in the reference legs remaining almost full. Other operator guidance regarding RPV water level indication will not be affected.

9 Resolution of Scram Discharge Volume Hi-Level Alarm Setting (SRI 97-001)

DESCRIPTION:

This Safety Review Item addresses a change of the Scram Discharge Volume (SDV) high-level alarm setpoint from a nominal value of 22 gallons to less than or equal to 28 gallons.

SUMMARY OF SAFETY EVALUATION:

The setpoint change addresses an error discovered in the volume calculation associated with the Scram Discharge Volume (SDV) setpoint. No physical changes to sensor location, the instrument loop, the alarm setpoint value (in inches) or the instrument volume occur. Therefore, a change of the SDV high-level alarm setpoint from 22 gallons to 28 gallons does not impact safety. The setpoint change is administrative in nature and does not affect the function of this instrument or any of the associated instrument loop interfaces. The new setpoint is within the operational and design criteria for this instrument.

The safety margin for the SDV is determined by the adequacy of the hydraulic coupling and the response of the scram instrumentation. The change does not impact the SDV safety performance. In determining whether a scram would go to completion under worst-case conditions, a degraded instrument air event coupled with maximum in-leakage is assumed to occur prior to the scram initiation. Under these conditions, the SDV fills quickly. The hydraulic coupling and scram instrumentation response are designed to assure scram performance without reliance on operator action.

10 Establishing "Restore and Maintain" Guidance for Drywell Temperature Limits in Emergency Operating Procedure C.5-1200, Part J. (SRI 97-007)

DESCRIPTION:

Primary containment control guideline, Operations Manual C.5-1200, Part J, was revised to allow "restore and maintain" actions when drywell atmospheric temperature exceeds 281°F. Mitigating actions would continue until the maximum ADS qualification temperature (335°F) or the drywell structure temperature limit (281°F wall temp) are challenged.

SUMMARY OF SAFETY EVALUATION:

These revisions are in agreement with the NRC approved BWROG Emergency Procedure Guidelines, Revision 4 (NEDO-31331). Revising the EOP logic to allow the drywell to exceed 281°F without requiring immediate emergency depressurization allows sequential action to restore drywell temperature, such that results of mitigating actions and their effects can be observed prior to initiating more aggressive action. Actions to restore drywell temperature would continue as long as it can be determined that containment integrity is not challenged and the drywell temperature remains below the maximum ADS qualification temperature. The operator actions and assumptions in the current Small Break Analysis (AE-083-0983) and the Power Rerate Analysis (GE-NE-T230071-1) remain valid.

11 Condensate Storage Tank Level Setpoint for HPCI/RCIC Suction Transfer and Related USAR Changes (SRI 97-008)

DESCRIPTION:

This Safety Review Item addresses: 1) An 8-inch increase in the setpoint for the Condensate Storage Tank (CST) HPCI/RCIC suction level transfer switches (LS 23-74 and 23-75), and 2) Approval of a temporary modification to cross-connect LS 23-74 and LS 23-75 when required during CST maintenance.

SUMMARY OF SAFETY EVALUATION:

The 8-inch setpoint change for the HPCI/RCIC low CST suction transfer function is made to accommodate differential pressure effects, vortexing, valve stroke time and level switch delay time. Up-to-date setpoint methodology is used. An increase in the setpoint is considered conservative since the transfer of HPCI and RCIC pump suctions to their safety related water source will occur earlier.

The temporary modification to cross-connect the CST level HPCI/RCIC suction transfer switches would have permitted extended maintenance outages of a CST and at the same time assure redundancy in the HPCI/RCIC level transfer logic. This temporary modification was not installed.

12 Clarification of Discrepancies of 4 KV Source Transfers Contained in USAR (SRI 97-009)

DESCRIPTION:

This Safety Review Item addresses discrepancies and ambiguities in USAR descriptions for the Auxiliary Power System and the Safeguards Diesel Generator System.

SUMMARY OF SAFETY EVALUATION:

The USAR revisions clarify information added during past revisions concerning EDG start logic and the logic involved with transfer between off-site sources. No new changes in plant equipment are involved. The USAR text changes involve elimination of a reference to a fast bus transfer to 1AR and revised discussion of common devices in the two divisions' diesel start and load logic, which were made extraneous by previous modifications of the EDG start circuitry.

13 Clarification of the Reactor and Radwaste Building Air Supply Description and the Reactor Building Ventilation Stack Description (SRI 97-010)

DESCRIPTION:

This SRI documents the evaluation made in support of changes to the USAR text so the system configuration and operation descriptions are consistent with the plant and how it is operated. The issue pertains to discrepancies between the practice of making seasonal filtering changes per Procedure 1151 (Winter Checklist) and the USAR wording, which implied the continual supply of filtered outside air, regardless of the season. Another discrepancy pertains to the description of the Reactor Building ventilation stack roof-top configuration.

SUMMARY OF SAFETY EVALUATION:

In December 1997, the NRC issued a Notice of Violation concerning a temporary change to Procedure 1151 which was intended to clarify clarified the conditions under which removal of the inlet filters is permitted during the winter season. In response, NSP committed to revise the USAR and Procedure 1151 to clarify filtering requirements and the basis for making seasonal filtering changes. The USAR changes do not require any associated Technical Specification changes.

14 Temporary Power to MCC 124 (SRI 97-011)

DESCRIPTION:

This SRI addresses the temporary supply of power to MCC-124 from power panel P-88 while the plant was shut down to install temporary supply feeders needed for MCCs B15 and B24 when a fault occurred in an underground portion of the normal feeders.

SUMMARY OF SAFETY EVALUATION:

MCC-124 is one of the two 480V motor control centers that supply loads to the off-gas stack and storage building. P-88 is a 480V power panel in the Site Administration Building that is fed from an off-site 12.5 KV feeder, which is not connected to the essential plant electrical system supply. There is no unreviewed safety questions associated with this temporary modification and no Technical Specification or USAR changes are required.

15 Temporary Installation of 480V Supply Feeders for MCCs B15 and B24 (SRI 97-012)

DESCRIPTION:

This SRI addresses the addition of temporary supply feeders for MCCs B15 and B24 until the corrective action for a fault in the underground portion of the permanent feeders is resolved.

SUMMARY OF SAFETY EVALUATION:

Exposure to the sun and the additional length of cable were evaluated. Physical damage to the temporary cables is prevented by placing cable underground where vehicles will be driven and placing barricades and labels at other locations. Training was provided to individuals involved with snow removal.

16 Operation With Asymmetric Control Rod Patterns (SRI 97-013)

DESCRIPTION:

This Safety Review Item provides justification for eliminating the requirement to insert and disable control rods in each of the other quarter core segments to offset the asymmetry resulting from Rod 14-07 being inserted and inoperable.

SUMMARY OF SAFETY EVALUATION:

As long as the requirements of the Banked Position Withdrawal Sequence (BPWS) regarding spatial distribution and the number of control rods are met, the acceptance criteria of the Control Rod Drop accident analysis are achieved with some asymmetry in control rod patterns. The NRC accepted report NEDO-21231 "Banked Position Withdrawal Sequence" allows up to eight inoperable, fully inserted, spatially distributed control rods. Sequences enforced by the Monticello Rod Worth Minimizer conform to the requirements of the generic BPWS as described in NEDO-21231.

GE report NEDE-32321 "3D Monicore (RD3L) Performance Evaluation Accuracy" confirms that the 3D-Monicore uncertainties, even with asymmetric control rod patterns, still satisfy the values assumed in the development of the SLCPR.

17 Scram Discharge Volume Capacity Change (SRI 97-015)

DESCRIPTION:

This Safety Review Item was prepared to confirm operability with a new computed scram discharge volume under worst-case scram conditions.

SUMMARY OF SAFETY EVALUATION:

As a result of a failure of a number of control rods to fully insert at Brown Ferry 3, the NRC issued IE Bulletin 80-17 on July 3, 1980. Then, in December 1980, the NRC distributed a generic SER regarding the BWR Scram Discharge system. It defined short-term actions considered necessary to assure continued safe operation of BWRs with inadequate Scram Discharge Volume (SDV) – Instrument Volume (IV) hydraulic coupling. Monticello installed and tested a new Scram Discharge Volume (Mod 81Z021) in late 1982 (NRC approval of the changes is reflected in License Amendment 11 dated December 9, 1982).

In 1988, a calculation (CA-88-018) was performed to determine SDV limit-switch settings and free volume calculations for Test 0006. It was determined that the scram discharge volume was smaller than that used in support of the 1982 modification. The difference was considered insignificant. In response to an October 21, 1993 QA Surveillance finding related to CA-88-018, field measurements were taken, and a new calculation (CA-97-151) was performed to more accurately determine the total volume of the SDV. The resulting volume was less than that previously computed but larger than the required 3.34 gal/drive.

The slightly smaller calculated total scram discharge volume results in slightly slower scram times. The effect of the slower scram times on the OLCPR was evaluated. It was determined that the Transient Initial Critical Power Ratio (CPR) for the worst-case plausible transient was less than the specified Operating Limit CPR in the Core Operating Limits Report (COLR).

18 Revision of USAR Section 8.8: Conduit, Cable Tray, Box and Cable Labeling (SRI 98-006)

DESCRIPTION:

This Safety Review Item involves an evaluation of USAR changes that clarify the labeling for conduits, cable trays, receptacle boxes and cables that require identification numbers. The SRI provides justification for removing the cable labeling criteria from the USAR and clarifies which conduits, cable trays, receptacle boxes and cables require labeling.

SUMMARY OF SAFETY EVALUATION:

The labeling guidance contained in the USAR was originally intended as an aid for field construction crews. It was incorporated into the FSAR at Amendment 21. There are no design, installation, operation or maintenance decisions based solely on cable tray identification numbers. Application of labels is adequately controlled by Procedure MWI-8-M-4.03, which provides specific requirements. Labels are an aid for field identification and do not perform any safety functions.

19 Startup Core Physics Testing (SRI 98-008, Rev 0)

DESCRIPTION:

This Safety Review Item addresses clarification of the applicability of a 1984 SRI (SRI-295) that deals with core physics testing to the technique used for the current cycle.

SUMMARY OF SAFETY EVALUATION:

The basis for issuing a new SRI to replace SRI 295 was a concern that the core physics testing activity for the current cycle had the potential of impacting the function of a system, structure or component described in the USAR. The new SRI has been reviewed by the corporate Nuclear Analysis Department (NAD) in accordance with NAD interface assessment I-98.065. NAD concurs that the method of core physics testing will provide the proper verification and/or comparisons necessary for startup core physics verification. It was also determined that there is no effect on the content of the USAR or the Technical Specifications.

20 Clarification of RPS Instrumentation and Wiring Environmental Qualification (EQ) Requirements (SRI 98-011)

DESCRIPTION:

This Safety Review Item addresses potentially confusing USAR discussions regarding the applicability of EQ requirements to Reactor Protection System instruments and wiring.

SUMMARY OF SAFETY EVALUATION:

No equipment safety classifications or qualification levels are revised. Removing the references to the EQ program in the USAR Section on RPS does not reduce any safety margin because the RPS is not required to be in the EQ program.

21 Change to Off-Gas Holdup System Bypass Requirement on High Recombiner Outlet Oxygen Concentration (SRI 98-012)

DESCRIPTION:

This Safety Review Item documents the basis for revising Operating Procedures and Alarm Response Procedures from initiation of Off-gas Holdup System bypass any time Recombiner outlet oxygen concentration exceeds 28% to initiation upon oxygen concentrations of greater than 28% for 10 minutes or greater than 44%.

SUMMARY OF SAFETY EVALUATION:

Results from testing in accordance with special Procedure 8889 provide an empirical basis for these changes. These procedural limits are self imposed and are not the basis for any plant Technical Specification or NRC safety evaluation associated with hydrogen water chemistry.

22 Diesel Engine Idler Gear Stub Shaft Assembly Replacement (EE 86-009)

DESCRIPTION:

This Engineering Evaluation was prepared to justify replacement of the engine idler gear stub-shaft assemblies on 11 and 12 Emergency Diesel Generators with the current design recommended in EMD Maintenance Instruction 9687. The replacement provides improved oil flow to the stub-shaft, idler gears and camshafts and increases the stub-shaft assembly strength.

SUMMARY OF SAFETY EVALUATION:

Testing performed upon completion of work proves operability. Therefore, the replacement of the engine idler gear stub-shaft assemblies does not create a possibility for an accident or malfunction of a different type than evaluated previously in the USAR or subsequent commitments, nor does it increase the probability or consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments. No USAR or Technical Specification changes are required.

23 Replacement of Diesel Engine Cylinder Head Assemblies (EDG 12) (EE 86-013)

DESCRIPTION:

This Engineering Evaluation was prepared to justify the replacement of the 12 Emergency Diesel Generator Circle-Two type cylinder head assemblies with Diamond-Five heads because the circle-two heads are no longer being manufactured.

SUMMARY OF SAFETY EVALUATION:

The diamond-five head is considered an upgrade because of the improved firing face strength and optimum cylinder cooling characteristics. All replacement parts meet or exceed the original specifications. Functional testing was performed by loading the diesel at 2500 KW for one hour. Replacement of the heads does not create a possibility for an accident or malfunction of a different type than evaluated previously in the USAR or subsequent commitments, nor does it increase the probability or consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments. No USAR or Technical Specification changes are required.

24 Replacement of Cylinder Head Assemblies on 11 Diesel Generator (EE 87-093)

DESCRIPTION:

This Engineering Evaluation was prepared to justify the replacement of the 11 Emergency Diesel Generator Circle-Two type cylinder head assemblies with Diamond-Five heads because the circle-two heads are no longer being manufactured.

SUMMARY OF SAFETY EVALUATION:

The diamond-five head is considered an upgrade because of the improved firing face strength and optimum cylinder cooling characteristics. All replacement parts meet or exceed the original specifications. Functional testing was performed by loading the diesel at 2500 KW for one hour. Replacement of the heads does not create a possibility for an accident or malfunction of a different type than evaluated previously in the USAR or subsequent commitments, nor does it increase the probability or consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments. No USAR or Technical Specification changes are required.

25 RHR System Pressure Upgrade (Mod 85M042, Addendum 2)

DESCRIPTION:

This modification increased the design values for the piping system within code allowables defined in ANSI B16.5, 1968 Edition. Addenda 2 addresses an issue arising from the NRC's 1997 System Operational Performance Inspection (Item E1.6) involving Section 10.2.4.3 wording changes incorporated with USAR Revision 12. It also covers the replacement of the instrument isolation valve for PS-10-118 with a new Whitney stainless steel valve.

SUMMARY OF SAFETY EVALUATION:

The intent of modification 85M042 was to increase the initiation pressure of the Shutdown Cooling Mode of RHR in order to reduce the critical path time to cold shutdown. To do this, the modification package called for increasing the design ratings of the RHR pump suction and discharge lines, raising the relief valve setpoints in the affected lines, replacing safety related instrumentation and removing the seal-in from the open direction of the Discharge to Waste Surge Tank Valve (MO-2407). However, increasing the setpoint of the RHR Shutdown Cooling Mode Interlocks was canceled prior to revising the USAR and the Technical Specifications. The Shutdown Cooling Mode is to be operated at the same temperatures and pressures as previously evaluated in the USAR. Therefore, this modification did not change the operational parameters of the Shutdown Cooling Mode but did increase the design values for the piping systems. The changes to the relief valve setpoints reflect these new design values to prevent spurious actuation. A hydrostatic test was performed in accordance with ASME Section XI requirements. The Safety function of the Discharge to Waste Surge Tank, MO-2407, was not changed.

The NRC's 1997 System Operational Performance Inspection (Item E1.6) cited this modification package for not providing adequate bases for determining that an Unreviewed Safety Question (USQ) did not exist. To resolve this USQ issue, Addendum 2 revises the USAR to the original operational values approved in the NRC's Safety Evaluation Report (SER) supporting License Amendment No. 22 dated February 2, 1984. The wording is changed to state that the maximum allowable pressure in the line is 215 psig, and the pressure relief setting is 150 psig as originally stated in the USAR. Per ANSI B16.5, the corresponding temperature for 215 psig is 281°F. The wording for design values is corrected to indicate 185 psig and 281°F.

26 Drywell Shell Interface Repair (Mod 92Q230)

DESCRIPTION:

This design change replaces the original Thiokol joint filler installed at the junction between the steel shell and the concrete bottom inside the drywell and the Pecora caulking installed at selected locations along this joint in 1987. The caulking installed in 1987 was at locations where drywell shell thickness measurements were taken as part of a Plant Life Extension program. The Thiokol and Pecora was removed and the surfaces cleaned of residue then the containment shell was painted. The joint was then filled with a non-shrinking grout to approximately ½ inch of the concrete surface, placing a closed cell polyethylene sheeting on the grout as bond breaker then applying silicone caulking.

SUMMARY OF SAFETY EVALUATION:

The effects of curing vapor on the SBGT charcoal filters were monitored to assure that Technical Specification limits on efficiency of the filters were not violated. Tests documenting the effectiveness of adhesion of the seal were completed prior to installation. This modification fulfills a commitment made on page 3 of NSP's May 11, 1987 response to NRC Generic Letter 87-05

27 SRV/MSIV Pneumatic System Upgrade (Mod 93Q305, Part C, Rev. 1 and 2)

DESCRIPTION:

This modification is associated with a Main Steam Isolation Valve (MSIV) Improvements project and involves modification of the Alternate Nitrogen System, which provides a safety related backup pneumatic supply to the Safety/Relief Valves (SRVs) and the inboard MSIVs.

SUMMARY OF SAFETY EVALUATION:

Modification 93Q305, Part C, Revision 1, addressed an environmental qualification issue (radiation exposure) associated with solenoid valves SV-2-71B and 71F. The corrective action involved replacing the valves with a model that used a metal rather than a Viton seat before the plant was returned to operation following the 1997 Emergency Core Cooling (ECCS) suction strainer outage. Revision 2 involves editorial changes to the associated evaluation to improve the description of consideration given to the ECCS function that Automatic Depressurization System (ADS) can provide under various failure scenarios to make it consistent with capability statements in USAR Section. USAR text changes associated with this modification have already been incorporated with Revision 13. No additional changes to the USAR were required.

28 Allow Silencing Emergency Diesel Generator (EDG) Circulating Oil Low Pressure Alarms (Mod 93Q415)

DESCRIPTION:

This modification replaced critical 11 and 12 EDG start/control circuit timing relays, adjusted the relay timing setpoint and/or range for certain non-critical timing relays and replaced the speed-sensing relay panels and associated magnetic pickup sensors. This design change also modified the wiring to the 11 and 12 EDG shutdown solenoids, modified the synchronizing motor control circuits to prevent the possibility of simultaneously energizing both the raise and lower motor windings, eliminated an undesirable interlock with the air start system compressors and modified the circulating oil and turbo circulating oil pump alarm circuits to allow the alarms to be silenced..

SUMMARY OF SAFETY EVALUATION:

This design change improves the operation of the 11 and 12 EDG control logic by replacing existing pneumatic timing relays with more accurate relays with solid state timers. In addition, timing setpoints and the allowable timing ranges on non-critical timing relays were adjusted. The speed sensing panels were also replaced since spare parts are no longer available for the original equipment.

Control wiring for the shutdown solenoids was modified to make it compatible with replacement shutdown solenoids. Replacement solenoids are polarity sensitive and would not have functioned properly with the original design wiring.

Changes were made to the synchronizing motor control circuits to ensure that the synchronizing motor does not receive simultaneous raise and lower signals which could potentially damage the motor.

The air start system compressor low oil level switch interlocks for the 11 and 12 EDG were removed. The switches have failed in the past. A level gauge (Site Glass) was installed and existing maintenance and inspection programs will ensure proper oil level to maintain operation of the compressors.

The original circulating and turbo circulating oil alarms could not be silenced and were in-service while the associated oil systems were not required. This design change modified the control logic to allow the alarms to be silenced and disabled when the these oil systems are not required.

29 Exciter Foundation Improvements (Mod 94Q135, Part B, C & D)

DESCRIPTION:

This multi-part modification deals with an upgrade of the main turbine. Part B consists of replacing the High Pressure (HP) turbine rotor and diaphragms, the LP turbine rotors, inner casings, diaphragms, shaft packing boxes and steam guides. Part B also includes increasing the capacity of the stator cooling water pumps and filters, increasing the isophase bus blower capacity, and replacing expansion joint XJ-1383 from the No. 8 steam extraction point to the E-13A intermediate pressure Feedwater heater. Part C involves strengthening and stiffening of the base and other support structure for the main generator exciter. Part D involves the additions of provisions for obtaining samples of stator cooling water for analysis.

SUMMARY OF SAFETY EVALUATION:

Part B modifications will enable improved plant output. Part C modifications strengthens and stiffen the support structure for the main exciter in order to reduce exciter bearing vibration levels that have gradually increased since the mid 1980s. Part D modifications enable monitoring for erosion/corrosion products in the stator cooling water. With one exception, the activities associated with the various parts of this modification either do not involve changes to the USAR and/or Technical Specifications or have already been addressed in previous USAR revisions. This USAR revision includes an updated Single Line Diagram - Station Connections drawing (NF-36175) which is associated with completion of Part B.

30 Reactor Feedwater Pump Recirc Valve Control (Mod 95Q055, Rev. 2)

DESCRIPTION:

Modification 95Q055 involved a Reactor Feedwater Pump (RFP) Recirculation Valve Control upgrade implemented in May 1996. Revision 2 completed implementation of software changes to activate the controllers.

SUMMARY OF SAFETY EVALUATION:

The recirculation controller replacement provides better system response through the use of anti-windup circuitry within the control loop to prevent low suction flow trips during feedwater flow transients. To further increase reliability, the time delay for the RFP recirculation valve close function after a RFP trip was reset from 50 seconds to greater than 10 seconds to allow for pump coast-down. Hand switches HS-1100 and 1103 on control room panel C-06 were replaced with switches that operate in the direction consistent with plant standards. The power supply for the RFP pump seal system was changed from a lighting panel to an Uninterruptible Power Supply source (panel Y-90) to increase system reliability.

31 Nitrogen Purge Line Low Temperature Isolation (Mod 95Q080)

DESCRIPTION:

This modification replaces the low and high-temperature switches, removes the nitrogen make-up high-flow alarm and incorporates an isolation on low nitrogen purge-line temperature. It also makes permanent a conduit sleeve to the Turbine Building.

SUMMARY OF SAFETY EVALUATION:

General Electric Service Information Letter SIL-402 discussed the discovery of cracking in the torus vent header at a BWR-4, which was attributed to brittle fracture caused by injection of cold nitrogen into the torus during inerting. This modification addresses the SIL-402 concern by providing a new temperature switch (TS-3276C) to the purge line temperature instrumentation loop that will isolate the Nitrogen purge line at a temperature setpoint of 40°F. During inerting operations, Procedure 2138 calls for Operators to maintain the purge line temperature at around 100°F and to stop Nitrogen flow if the temperature cannot be maintained above 60°F. The low-temperature isolation feature will provide additional protection against the inadvertent flow of cold nitrogen into the drywell or torus.

32 Intake Dredge Settling Basin (Mod 95Q220)

DESCRIPTION:

This modification involves the construction of a non-safety related concrete-lined 21'x75' by 40-inch deep, three-chamber settling basin off the southeast corner of Cooling Tower No. 12.

SUMMARY OF SAFETY EVALUATION:

The new settling basin is used during periodic dredging of the plant intake canal, which is performed to prevent river silt from clogging the main intake pumps. Dredge pump effluent is routed through the three basins to the No. 12 Cooling Tower catch basin. The pumping rate provides approximately a 30-minute hold-up within the basin for settling of suspended solids. A ramp entrance is provided to enable a front-end loader to clean out the accumulated sediment after the basin is drained.

33 Modify Drain Line SW32-4"-JF to Eliminate Freezing (Mod 95Q265)

DESCRIPTION:

This modification changes the routing of the Residual Heat Removal Service Water (RHRSW) motor cooler discharge flow from a common drain header (SW32-4"-JF) connection, which was subject to freezing, to the intake structure floor drain. This project also included an operability evaluation of the RHRSW motor cooler supply lines to identify modifications required to support the Seismic Class I classification of these lines.

SUMMARY OF SAFETY EVALUATION:

Changing the discharge flow from the outlet of the RHRSW motor coolers to the floor drains near the RHRSW pumps eliminates the possibility of flow blockage due to a frozen drain header. This change also separates the RHRSW divisions on the effluent of the motor coolers. Motor cooler supply lines found to be classified as Seismic Class II were upgraded to Class I. By removing the Safety/QA related RHRSW motor cooler drain line connections from SW32-4"-JF, this line could be reclassified as non-safety related.

34 Reactor Building Roof Superstructure (Mod 96Q005)

DESCRIPTION:

This modification involves reinforcing the Reactor Building superstructure to increase the load carrying capability. Base plates were strengthened by the addition of stiffening plates and various members and connections in the trusses were strengthened by added weld and/or addition of reinforcing plates.

SUMMARY OF SAFETY EVALUATION:

This modification was made to assure that the Reactor Building steel superstructure will not generate any missiles large enough to pose a hazard to the spent fuel in the fuel pool as a result of a design basis tornado wind of 300 MPH. Although the Reactor Building superstructure is not a Class I structure, it is designed for the tornado load. The physical modifications bring the design into compliance with the evaluation performed for the original design basis.

35 Control Room Dose Reduction (Mod 96Q015, Part B and D)

DESCRIPTION:

The overall objective of modification 96Q015 is to assure a path for leakage past the MSIVs to the condenser. The approach is to verify the seismic capability of the condenser and one of the paths from the outboard MSIVs to the condenser. Part B of the design change corrects certain deficiencies that were identified during the seismic verification effort.

SUMMARY OF SAFETY EVALUATION:

The deficiencies corrected under Part B involved replacement of carbon steel piping in the scope of this project (which had been identified as having through-wall or wall-thinning erosion) with stainless steel piping. The deficiencies corrected under Part D are the supports on the Condensate backwash receiving tank (Tank T-33) and on line OG6-10"-HC (Tank H-33 vent line).

36 Resolution of Seismic Qualification Users Group (SQUG) Outliers (Mod 96Q035)

DESCRIPTION:

This modification addresses the outliers identified in "NRC Unreviewed Safety Issue (USI) A-46 Seismic Evaluation Report - Monticello Nuclear Generating Plant" sent to the NRC in November 1995

SUMMARY OF SAFETY EVALUATION:

An "outlier" is an item of equipment that did not comply with all of the screening guidelines provided in the Generic Implementation Procedure, revision 2, as corrected on February 14, 1992 (GIP-2). In November 1995, NSP sent a report to the NRC summarizing the results of the seismic review that was performed to address the concerns of USI A-36. Table 7-1 listed the outliers. The modifications performed assure that the identified outliers are in compliance with the GIP criteria and are, therefore, deemed to be seismic qualified.

37 Selected Motor Operated Valve (MOV) Upgrades - Final Turnover (Mod 96Q045)

DESCRIPTION:

This modification involves upgrades to selected MOVs. A number of the changes were implemented during the 1996 refueling outage and addressed in USAR revision 14. The remaining change was completed during the 1998 refueling outage, including the drilling of holes in the upstream disc for MO-2006 to address NRC Generic Letter 95-07 on pressure locking and thermal binding.

SUMMARY OF SAFETY EVALUATION:

The drilling of holes in the RHR pump side disc of MO-2006 and MO-2007, together with an earlier NRC exemption from 10CFR50, Appendix J, enable removal of these valves from the list of containment penetration isolation valves requiring Type C Appendix J tests. The hole in the disc allows water from the RHR pump to pressurize the bonnet area thus eliminating this potential fission product leakage path.

38 Relocation of "A" RHR Conductivity Cell (Mod 96Q055)

DESCRIPTION:

This modification moves the RHR-A conductivity sample from the heat exchanger bypass line to the existing Post Accident Sampling System (PASS) tap at the heat exchanger effluent.

SUMMARY OF SAFETY EVALUATION:

Having the sample taken from a location immediately downstream of the heat exchanger makes detection of a tube leak easier.

39 Reactor Water Cleanup (RWCU) Pump Improvement (Mod 96Q130, Rev. 1)

DESCRIPTION:

This modification involves the removal of the cooling lines from Reactor Building Closed Cooling Water (RBCCW) system to the RWCU pumps that supply the seal cooler, the stuffing box and the bearing cooler. It also involves removal of the heat exchanger, installation of cartridge seals, and removes TIS-12-89A & B (high temperature switch and pump trip logic). A new line ties into the pump vent line to supply flushing/cooling water to the new seals. The seal for RWCU Pump P-204A (#11) and the RBCCW cooling lines to the pump were removed. RBCCW cooling lines to RWCU pump P-204B (#12) will be removed when this pump seal needs to be replaced.

SUMMARY OF SAFETY EVALUATION:

This design change achieves reduced personnel radiation exposure for future RWCU pump maintenance activities by reducing the time required to remove and reinstall unnecessary cooling lines and simplifying the seal design. RWCU piping design and installation is in accordance with ANSI B31.1, 1977 Edition with all addenda through Winter 1978. Welding was completed by welders qualified in accordance with ASME Code Section IX using qualified weld procedures from the NSP Weld Manual. NDE consistent with 5828-M-200 "Testing and Inspection Schedule" was performed during initial service leak checks to assure integrity of the welds. The RWCU pump control circuit was tested for proper operation following the removal of the high-temperature trip.

40 Hydrogen Water Chemistry (HWC) Improvements (Mod 96Q145, Rev. 1)

DESCRIPTION:

Modification 96Q145 involved the installation of a billing meter on the hydrogen supply line and a reclaimer compressor on the liquid hydrogen storage tank vents. This revision reduced the project scope by eliminating a planned software upgrade for the hydrogen controllers.

SUMMARY OF SAFETY EVALUATION:

After the project was approved, it was determined that the testing of the software upgrade required a Reactor Feedwater Pump (RFP) trip. It was determined that the benefits of the software upgrade are not enough to offset the cost of a RFP trip.

41 RHR SW Air Vent Valve Modification (Mod 96Q150, Addendum 1, Rev 1)

DESCRIPTION:

This modification involves actions taken to correct operational problems (failure to properly seat) associated with air valve AV-3147, which is installed on the discharge from the 11 RHR Service Water Pump.

SUMMARY OF SAFETY EVALUATION:

An air vent valve was installed on the discharge of each RHR Service Water pump to vent air from the pump column and piping upstream of the pump discharge check valve when the pump starts. This valve also allows water to drain from the piping and pump column after the pump is shut down. Modification 96Q150 was initiated to address operational problems (failure to seat) with air vent valve AV-3147. As an interim measure, an orifice plate was installed in the flanged connection at the inlet to AV-3147 to allow the air to escape but throttle the flow of water so the valve seats with less impact. After determining that the orifice did not resolve the seating problem, Addendum 1 was prepared to address the removal of the orifice and installation of a surge check valve for the air vent valve on each of the four RHR Service Water pumps. Revision 1 to this Addendum addresses a gasket seating issue relative to the flanges for the new surge check valves.

The surge check valves have been evaluated for Class 1 loads. The effect of the branch connection on the RHR Service Water piping was also evaluated to verify that the piping stresses are still within code allowables.

42 Control Room and Warehouse Network Installation (Mod 96Q155)

DESCRIPTION:

New optical computer network links are established between the computer room and the control room and the warehouse 1 & 2 buildings.

SUMMARY OF SAFETY EVALUATION:

These computer links have no effect on other plant systems.

43 Single Rod Scram Timing Panel Improvements (Mod 96Q165)

DESCRIPTION:

This modification involves replacing the electromagnetic relay used for the auto and single rod scram timing marker inputs to the Rod Worth Minimizer with solid state relays. The relays associated with the rod drift alarm reset circuit were replaced and indicating type fuses associated with the auto scram timing marker circuits were replaced with safety related, fully coordinated, non-indicating type fuses. The abandoned scram timing recorder equipment located in Cable Spreading Room Panel C-28 and test jacks and switches located on Control Room Panel C-16, which are no longer required for the scram timing, were removed.

SUMMARY OF SAFETY EVALUATION:

This modification is associated with the commitment in NSP's May 10, 1996 letter to monitor CRD Scram Solenoid Pilot Valves (SSPV) with Viton internals in accordance with BWR Owners Regulator Response Group recommendations. Achieving the monitoring recommended by the BWROG requires temporary installation of testing equipment in the Reactor Protective System (RPS) panels. This modification changes the single rod scram insertion timing circuits to eliminate the need for temporary testing equipment and wiring changes. The changes also improve the accuracy of the auto scram timing marker circuit used to determine the rod insertion times following an auto scram.

44 Replace Relief Valves RV-2993, RV-1745 and RV-1746 (Mod 96Q180)

DESCRIPTION:

This modification replaces relief valves RV-1745 and 1746 on the discharge of the Core Spray System with valves having setpoints increased from 375 to 500 psig. It also changes the pressure rating of the cooling water supply to the Reactor Core Isolation Cooling (RCIC) turbine lube oil cooler to reflect the 75 psig setpoint of RV-2097 and replaces RV-2993 with a valve having a setpoint of 165 psig.

SUMMARY OF SAFETY EVALUATION:

Neither the function of the relief valves nor the pressure rating of the piping was changed. However, the replacement relief valves are upgraded to ASME Section VIII code stamped components.

45 Correction of CRV-EFT Timing Relay Seal Problem (Mod 96Q185)

DESCRIPTION:

This modification moves the supply and exhaust fan trip functions to the same timing relay that provides the seal-in function so the low-flow trip circuit seals in at the same time that the Control Room Ventilation - Emergency Filtration Train (CRV-EFT) system supply and exhaust fans are tripped. The CRV-EFT refrigerant compressor supply breaker instantaneous setting was also increased to enhance system reliability.

SUMMARY OF SAFETY EVALUATION:

The CRV-EFT supply and exhaust fans are designed to trip if low flow is detected in the exhaust fan duct. This design change modifies both trains of the CRV-EFT system low flow trip circuits so the supply fan permissive, the exhaust fan permissive and the low flow trip seal-in functions are all performed by one timing relay. The design change also revises both trains of the CRV-EFT refrigeration compressor supply breaker instantaneous settings. These changes were made to enhance system reliability.

46 Control Room Kitchen Replacement (Mod 96Q200)

DESCRIPTION:

This modification replaces an existing control room kitchen unit with a new prefabricated kitchen unit.

SUMMARY OF SAFETY EVALUATION:

The small kitchen provided in the control room allows the control room operations staff to prepare meals without having to leave the control room. The replacement unit consisting of prefabricated metal cabinets with range, refrigerator, microwave and lighted hood fan were sized to fit into the existing kitchen space without any major wall remodeling.

47 Replace Pump P-90 and Hand-Switches HS-2509, HS-2510, HS-2589 and HS-2590 (Mod 97Q000)

DESCRIPTION:

This modification installs a larger "Trash" pump to replace the Condenser Waterbox Pump P-90, removes strainer YS-4140 with associated valves CW-48, CW-48-1 and CW-48-2, and replaces the High Pressure Condenser Outlet Air Vent valve hand-switches HS-2589, HS-2590, HS-2509 and HS-2510 with auto/open, two-position, maintained contact switches.

SUMMARY OF SAFETY EVALUATION:

The larger trash pump and associated piping changes will handle 2-inch solids and reduce the amount of time required to pump down the Condenser water boxes during partial Condenser cleaning. The replacement switches will eliminate the need to use jumpers in C06.

48 Condensate Pump Replacement (Mod 97Q010, Rev 1)

DESCRIPTION:

This modification involves the procurement and installation of two new condensate pumps and one new condensate pump motor. The existing pumps and the current #12 Condensate Pump motor were removed and retained as spares.

SUMMARY OF SAFETY EVALUATION:

The installation of new condensate pumps affect the Feedwater runout flow, which is an input to the Transient Protection Parameters (TRAPP) form. A bounding feedwater runout flow is used in the analysis to determine the Core Operating Limits. The new condensate pumps do not exceed this bounding feedwater runout flow. The installed performance data of the new condensate pumps is used to update the runout analysis calculation and TRAPP form as required. Replacement of the General Electric motor on the #12 Condensate Pump with a Siemens motor includes removal of the bearing oil cooling pump and heat exchanger. The bearing oil cooling system is being retained with the General Electric motor as a spare.

The replacement condensate pumps increase the maximum runout flow. The bounding runout flow value used in the Core Licensing analysis exceeds that available from the new condensate pumps. The new condensate pumps are of equal or better quality than the original pumps and provide considerable gains in Net Positive Suction Head (NPSH) and efficiency.

49 Reactor Building Catwalk and Access Platform (Mod 97Q015)

DESCRIPTION:

This modification installed: 1) a catwalk access to the fire detectors on the ceiling of the refueling floor in the Reactor Building; 2) an access platform for the Reactor Building Ventilation Wide Range Gas Monitor (WRGM) Process Flow Probes on the north side of the Refueling Floor; and 3) a permanent steel platform in the torus room to access the valves required to perform monthly Torus-Drywell Vacuum Breaker Test #0143.

SUMMARY OF SAFETY EVALUATION:

The platform for the Reactor Building fire detectors is attached to the superstructure and designed to remain intact during a seismic event. The platform over the HVAC duct provides access to the WRGM probes for calibration without standing on the ductwork. The platform added to the torus room is a Class II structure that is designed to Class I criteria and anchored to the compartment wall. Access is from the existing ladder from the torus catwalk to the basement floor.

50 Piping and Equipment Modifications to Support Power Rerate (Mod 97Q020, Part A & B)

DESCRIPTION:

This design change consists of plant piping and equipment evaluations to assure that they will continue to meet their design basis when considering the effects of an increase in reactor thermal power to 1775 MWt. The scope of the evaluations includes Seismic Class I and II, safety and non-safety related, QA and non-QA related system components. Fire protection systems are not included.

SUMMARY OF SAFETY EVALUATION:

The evaluations concluded that the current design parameters for the following system piping and equipment are adequate for operation at 1775 MWt conditions:

- Recirculation system
- Reactor Water Cleanup system
- CRD insert and withdrawal lines
- CRD Scram discharge volume
- CRD hydraulic system
- Emergency Diesel Generator Service Water system
- Gaseous Waste systems
- Reactor Head Vent and Head Seal Leak Detection
- Reactor Vessel Instrumentation

Part A of Modification 97Q020 addresses Emergency Service Water and Torus attached piping.

Part B addresses:

- Main Steam piping
- RHR piping and pumps
- HPCI piping, pump flow rate capability and operability
- RCIC piping, pumps and operability
- SRV discharge lines
- Feedwater/Condensate piping, pumps and pressure vessels
- Extraction steam, Feedwater heater drains and miscellaneous steam piping
- RHR Service Water piping
- RHR heat exchangers
- Reactor Building Closed Cooling Water (RBCCW) piping
- Circulating Water system piping
- CGCS piping and skids
- Service Water and Fuel Pool Cooling

The main steam piping down stream from the turbine stop valves, the condensate, feedwater, extraction steam, feedwater heater drains and miscellaneous steam piping was evaluated for operation up to 1818 MWt because the HP turbine steam path is designed for a valve wide open steam flow for a core thermal power of 1818 MWt. For the other evaluations, a thermal power level of 1880 MWt was used.

Physical modifications resulting from these evaluations include the following:

- Modification of top and bottom flanges of steel beams on each RHR Heat exchanger
- Addition of stiffener plates to an existing support base plate for the RHR A division heat exchanger
- Modification of support HDH-32 on the drain line from Feedwater heater E-12A to drain cooler DC-12A by replacing the spring can with a rigid support

- Attaching new Code Name Plates on affected Feedwater heaters and drain coolers
- Adjusting spring can pre-load on TWH-62 on RHR loop A discharge pipe
- Remove support FPWH-80 from the fuel pool system return line from the filter/demineralizes

None of the physical changes affect any descriptive discussions in the USAR. Line designation tables (not part of the USAR) are updated to reflect the revised design temperatures. There are changes to the USAR associated with the Part B scope. Those changes to the USAR text and tables that pertain to the system design capability are being included in this general USAR update (USAR Revision 16). Changes that pertain to operating parameters are being withheld until the USAR revision that is to be submitted after the power ascension testing associated with NRC approval of the power rerate program.

51 Feedwater Heater Vent Piping Replacement (Mod 97Q030)

DESCRIPTION:

This modification replaces portions of the six carbon steel (pipe class HB) vent lines coming off the shell sides of the E-11A&B, E-12A&B and E-13A&B Feedwater Heaters with erosion-resistant stainless steel pipe (pipe class HCD). The new lines follow the same routing using the same pipe supports.

SUMMARY OF SAFETY EVALUATION:

The feedwater heater vent lines carry non-condensables from the shell sides of the heaters to the main condensers, and are subject to erosion due to the high moisture content. The replacement piping chosen is Schedule 40 ASTM A312 TP 304L stainless steel due to its greater resistance to steam erosion. The routing was not changed in order to use existing hangers and supports. One rod hanger was modified to allow unrestricted expansion in the vertical direction. The modification reduces the possibility of steam leaks inside the condenser room.

52 Drywell Atmosphere Cooling 4-Fan Pressure Differential Control Removal (Mod 97Q035)

DESCRIPTION:

This modification removed the pressure differential control and disconnected the fan low flow trips associated with the modulated control feature for the four drywell cooling units. The differential control switches for the 3-fan mode were also upgraded to a newer model because the original component was obsolete and no longer manufactured.

SUMMARY OF SAFETY EVALUATION:

The installation of new cooling coils in the four drywell cooling units (Mod 82M079) eliminated the need for modulated control of the inlet dampers when in the 4-fan operating mode (Mode 8) of drywell cooling. Removal of the 4-fan differential pressure control switches eliminates the need to perform periodic calibration of these devices. Calibration was difficult, and the devices were located in an area of considerable radiation.

Entering mode 8 operation of drywell cooling will now require additional operator action to open the inlet damper for the fourth fan and adjustment of the other operating fan units for desired flow.

53 Primary Containment Penetration X-16B Bellows Replacement (Mod 97Q050, Rev. 1)

DESCRIPTION:

This modification involves replacement of the metallic expansion bellows at the primary containment penetration for the Core Spray A discharge piping (Penetration X-16B). Revision 1 incorporates the 1980 Edition of Section III, Subsection NE of the ASME code. Use of Inconel was first recognized in the 1980 Edition Summer 1982 Addenda. Inconel was the preferred material for the expansion joint replacement.

SUMMARY OF SAFETY EVALUATION:

Although there are some configuration differences, the new bellows is designed to meet or exceed the original codes of construction. The replacement bellows includes radial separation between the two plies. This addresses leak testing difficulties and provides more accurate Local Leak Rate Testing results. Inconel was chosen over stainless steel for its higher resistance to Intergranular Stress Corrosion Cracking (IGSCC) and fatigue.

The design pressure for the new bellows was increased from 56 to 62 psig to address changes to ASME Section III. The original bellows was designed in accordance with the 1965 edition of the code. The new design is in accordance with the 1977 edition of ASME Section III, Subsection NE with Addenda through Winter 1978. Replacement activities conform to the 1986 edition of ASME Section XI.

54 Secondary Containment Trip Of The Radwaste Exhaust Fans (Mod 97Q055)

DESCRIPTION:

This design change adds divisionally redundant trips of Exhaust Fans V-FU-1 and V-FU-2 during a Secondary Containment (SCT) isolation. This design change also installs a non-safety related trip of the Turbine Building supply fan V-MZ-4 on SCT isolation.

SUMMARY OF SAFETY EVALUATION:

This design change will reduce the potential for leakage from SCT to the Radwaste Building when Standby Gas Treatment System (SGTS) is operating during a SCT isolation. Tripping V-FU-1 and 2 fans will assure that the Radwaste Building pressure is not negative with respect to the Reactor Building. Installing a SCT isolation trip of the Radwaste exhaust fans will replace operator action to turn the fans off within Operations Manual Abnormal Procedure C.4-B.4.1.B, Primary Containment Group 11 Isolation.

Tripping the Turbine Building Supply fan V-MZ-4 will improve performance of the SCT system and allow the fan to be restored to service without the concern of pressurizing the Turbine Building during a SCT isolation.

55 Modify Drywell Penetration Flued Head Anchors (Mod 97Q060)

DESCRIPTION:

This modification involves changes made to the supports for the flued head penetrations X-11 and X-16A. The combined support for penetrations X-12 and X-13B was also re-analyzed to verify compliance with code.

SUMMARY OF SAFETY EVALUATION:

During the design basis reconstitution program, some inconsistencies were found between the USAR identified design loading conditions and the methods used to evaluate the corresponding stresses in containment penetrations and associated supports compared to the loading conditions and methods documented in design specifications. This modification addresses the supports for flued head penetrations X-11, X-12/X-13B, and X-16A. The X-11 and X-16A supports were modified by adding extra bracing specifically designed to withstand the torsional loads postulated in the re-analysis performed to address these inconsistencies. The combined support for X-12 and X-13B was re-analyzed using the as-built configuration.

56 Reactor Water Cleanup (RWCU) Line Break Isolation Logic (Mod 97Q065)

DESCRIPTION:

This modification adds one-out-of-two-twice high RWCU room temperature and RWCU system flow signals to the existing Primary Containment System Group 3 logic to automatically isolate a RWCU High Energy Line Break (HELB). In addition, a rod hanger between the X-14 penetration and the first anchor in the RWCU room not shown on any drawing was removed.

SUMMARY OF SAFETY EVALUATION:

During reviews of HELB analyses for the power rerate program, it was determined that the calculated mass flux from a postulated RWCU break within the Reactor Building was higher than assumed in the original plant licensing evaluations. In addition, it was found that rapid automatic isolation of the RWCU break on a low reactor water level will only occur when the plant is operating at near full power. The limiting RWCU HELB is where automatic isolation will not occur within the first 10 minutes. This occurs at reactor power levels less than 77%. An evaluation to support continued operation determined that the RWCU HELB analysis is still bounded by the Main Steam Line break radiological analysis as long as the reactor water dose equivalent I-131 is 0.25 μCi/g or less. A Licensee Event

Report (LER) was transmitted to the NRC on September 16, 1996. NSP committed to investigate additional automatic RWCU isolations, revise the USAR to reflect changes associated with this issue, and submit a License Amendment Request (LAR) to revise the Technical Specifications to reflect a reactor coolant dose equivalent Iodine limit of 0.25 μCi/g.

USAR Revision 14 removed inconsistencies relative to the current HELB requirements, clarified the definition of safe shutdown in the USAR HELB analysis and expanded the applicability of HELB analysis to all plant operating modes. License Amendment 101, issued August 28, 1998, approved the revised dose equivalent Iodine. USAR changes to reflect this approval will be incorporated in the next USAR revision.

Modification 97Q065 provides automatic isolation of MO-2397, MO-2398, and MO-2399 on high RWCU room temperature and high RWCU system flow. Installation was completed during the 1998 refueling outage. The automatic isolation added by this modification will limit the potential radiological consequences from a RWCU HELB to less than that postulated for the Main Steam Line break accident. This is consistent with the original licensing basis.

57 Turbine Lube Oil Purifier Modification (Mod 97Q075)

DESCRIPTION:

This modification replaced the main turbine oil purifier system logic controller with instrumentation to automatically control the speed of the turbine lube oil pump over the lube oil operating temperature region (70 - 100°F).

SUMMARY OF SAFETY EVALUATION:

When the turbine lube oil purifier failed in 1996, a GE adjustable speed motor drive with manual speed adjustment control was installed, and the system logic controller was bypassed so operation could be resumed in manual mode. This modification restores the automatic lube oil pump speed control function.

58 LPCI Loop Select Timer Replacement

DESCRIPTION:

This modification replaces four Agastat E7000 series Electro-pneumatic time delay relays in the LPCI loop select logic with Agastat type ETR series solid state time delay relays.

SUMMARY OF SAFETY EVALUATION:

This modification addresses potential concerns regarding repeatability accuracy identified in NRC Information Notice 92-077 "Questionable Selection and Review of Electro-pneumatic Relays for Certain Applications". The LPCI loop select functions for these relays have not been changed or altered. The relay time delay settings for each relay will remain the same.

59 Steam Jet Air Ejector Off-Gas Discharge Pipe Drain (Mod 97Q095)

DESCRIPTION:

This modification involves the installation of a drain in a section of piping (GOLE-118-6"-HN2C) between the Steam Jet Air Ejector (SJAE) discharge and the 24" delay line.

SUMMARY OF SAFETY EVALUATION:

During investigation of an Off-Gas flow spiking problem, a 20-foot section of horizontal 6-inch piping was found half full of water using ultrasonic measurement techniques. Normal flow through the line is wet off-gas. Other sections of off-gas piping are sloped to allow draining of Condensate. It was suspected that water in the line surged from end to end when the SJAE discharge was suddenly changed due to condensate valve cycling. To correct this problem, a small drain was installed in the affected line.

60 MO-2373/2374 MOV Replacement (Mod 97Q110)

DESCRIPTION:

This modification involves replacement of the valves and actuators for Main Steam Line Drain inboard and outboard isolation valves (MO-2373 and MO-2374).

SUMMARY OF SAFETY EVALUATION:

In order to meet as-left Appendix J leak rate requirements during the 1994 refueling outage, the thrust output of the Limitorque actuators for MO-2373 and MO-2374 was increased such that the maximum allowable torque ratings were exceeded. An evaluation of this condition determined that it was acceptable for a period of time, then gear replacement was required. Replacing these valves and their actuators improves leak rate performance and resolve the actuator over-stress concerns. The higher short-term total load for MO-2373 is still less than the diesel generator unit load rating.

61 Vessel Head Vent Line Modification (Mod 97Q120)

DESCRIPTION:

This design change modifies the vessel head vent line (V15-2"-ED) to assure that the original code-allowable stress limits are met.

SUMMARY OF SAFETY EVALUATION:

The purpose of this modification is to restore stress levels to within code-allowable limits using current analysis methods.

62 Vacuum Breaker AD-2382A Modification (Mod 97Q125)

DESCRIPTION:

This design change involves the installation of a spacer between the disc arm and the disc nut washer to reduce the axial movement of the disc post on the Atwood & Morrill drywell-to-torus (wetwell) vacuum breaker AO-2382A.

SUMMARY OF SAFETY EVALUATION:

The purpose of this design change is to improve the valve seating alignment of the vacuum breaker. Report NSP-53-107 "Dynamic Analysis of Wetwell-to-Drywell Vacuum Breakers" documents the ability to withstand design loading due to chugging and load combinations outlined in the Mark I Program Structural Acceptance Criteria. An evaluation has been performed to verify that the addition of a spacer does not adversely affect compliance with these acceptance criteria, the counter weight (weight or arm), or the structural capability of the vacuum breaker.

63 Sparing of Breaker 152-403 (Mod 97Q135)

DESCRIPTION:

This modification disconnects the cables from Breaker 152-403 to former substation auxiliary power transformer X41, disables controls and indicators associated with the breaker and abandons Breaker 152-403 in place as a spare. Associated with this modification is the installation of an alternate source of auxiliary power to the Monticello substation.

SUMMARY OF SAFETY EVALUATION:

The 4 KV cubicle 152-403 was utilized for the feeder to substation auxiliary power transformer X-41. X-41 stepped down the 4 KV to 240/120 V for use as a power supply to substation auxiliary components. An alternate source for substation auxiliary power is from plant Bus 13 via the 4 KV cubicle 152-303 and transformer X-31. On January 18, 1998, the underground cable interconnecting X-41 and cubicle 152-403 faulted to ground. The breaker in cubicle 152-403 was racked to the test position to isolate the faulted cable from the plant 4KV system.

An evaluation of repair options determined that it would not be prudent to attempt replacement of the failed cable due to its close proximity to other critical cables buried in the same trench. In addition, NSP has a program in place to improve the reliability of the redundant plant substation auxiliary power systems by establishing one source from the plant and the other from the substation transmission system components. Based on these factors, a decision was made to abandon the faulted feeder to X-41 and install a new source.

Associated with this modification is Substation Project E #97ES01, which provides a second source of auxiliary power to Monticello substation This project replaced the X-41 transformer with a new 13.8 KV to 240/120 V transformer fed from the tertiary winding of the Number 10 transformer via a newly installed fused disconnect switch. To comply with the standard labeling practices in other NSP substations, the newly installed transformer is labeled "Substation Auxiliary Transformer B" and the existing transformer X-31 is re-labeled "Substation Auxiliary Transformer A".

64 1R and Instrument AC Transformer TAP Changes (Mod 97Q140)

DESCRIPTION:

This modification revises the 1R supply transformer no-load tap setting to reduce the transformer ratio by 2.5% (change from 115,000/4160 to 112,125/4160 setting) thus increasing the secondary supply voltages with respect to a given primary voltage. This project also includes associated changes to 480/120 transformer tap settings required to maintain desired Instrument AC control voltage ranges.

SUMMARY OF SAFETY EVALUATION:

The transformer tap changes improve the overall reliability of off-site power to the plant and provides margin for future plant load additions.

65 ECCS Suction Strainer Modification (Mod 97Q170)

DESCRIPTION:

The scope of this project includes removal of the old ECCS suction strainers, installation of new ECCS suction strainer assemblies and their suppo ts, and the required reinforcement of the torus and the torus ring girder. The torus access hatch in the 935-ft elevation of the Reactor Building was enlarged to allow the new strainers to be lowered into the torus manway.

SUMMARY OF SAFETY EVALUATION:

The new strainers meet the original design requirements and will not change any of the USAR assumptions or criteria or any of the strainer design criteria. Strainer head loss was determined by analysis and testing performed by PCI. A multi-system performance test was performed following the installation of the new strainers. The effect of the new strainer assemblies on the Mark I torus design has been evaluated, and local reinforcement of the shell and ring girder was added to maintain stresses in the torus below code allowables consistent with the Monticello Mark I Plant Unique Analysis Report (PUAR). The new hatch cover has been designed as a fire barrier, a HELB barrier and for internal flooding.

By letter dated August 12, 1997, NSP committed to change the Monticello suppression pool technical specification limits to correspond to the associated instrumentation limits under the Improved Technical Specification (ITS) program. The maximum water volume limit of 72,910 cu-ft will be changed to recognize the corresponding level change due to approximately 69 cu-ft of displaced water in order to maintain the current maximum downcomer submergence.

66 Standby Gas Treatment System (SGTS) Grounding (Mod 97Q180)

DESCRIPTION:

This modification installed a solid building ground grid from the Reactor Building to the major electrical equipment and panels inside the SGTS room.

SUMMARY OF SAFETY EVALUATION:

Completion of this work addresses the actions required to address lightning susceptibility concerns and potential problems associated with instrument signal shielding conventions.

67 4KV Essential Bus Transfer Logic Testability Improvement (Mod 97Q195)

DESCRIPTION:

Design Change 97Q195 makes minor wiring changes to the Division I and Division II 4KV loss of essential bus voltage logic circuits and adds indicating lights in 4.16KV essential bus cubicles 152-510 (P-61 - Main Turbine Auxiliary Oil Supply Pump 4KV Supply) and 152-601 (16 Bus Cross-Tie 4KV Supply).

SUMMARY OF SAFETY EVALUATION:

This Design Change simplifies the monthly testing of the 4.16KV Essential Bus Transfer logic performed under Procedure 0301 by eliminating the need to use a multi-meter to monitor operation of degraded voltage relays 127-5Y, 127-5Z, 127-6Y and 127-6Z. Safety related indicating lights were installed to provide confirmation that the relays operate.

68 Replacement of Root Valve for PT-1176 (Mod 97Q210)

DESCRIPTION:

A bypass tag was hung on the root valve for PT-1176 that provides "B" steam line pressure to the control room and low pressure isolation signal to the air ejector suction valves after temporary repairs (Furmanite) were made to stop a packing leak. This modification replaced the root valve for PT-1176 with a type that has a metal diaphragm. Associated with this modification was the removal of a one-foot long branch line immediately downstream of the root valve that had been capped off and abandoned in place.

SUMMARY OF SAFETY EVALUATION:

Components installed per this design change meet all applicable code requirements and are designed and aptly suited for the conditions under which they will be operated. The replacement valve provides a component that is less prone to packing and bonnet leaks.

The original design and fabrication code was USAS B31.1.0-1967. The code used for the modification was ANSI B31.1-1977 with Addenda through Winter 1977 in accordance with NSP specification NPD-M-38.

69 Cross Tie Floor Drain Collector Tank (Mod 97Q215)

DESCRIPTION:

This modification installs a tee and two gate valves in the line from the P-39 pump to tank T-24 to allow flow to be routed to either T-24 or T-26 or both. A spectacle flange and blind flange tee is also installed in the line from tank T-22 to T-24 to allow a temporary connection for diverting effluent from T-22 to T-26. A new penetration is drilled between the Radwaste Building and the Reactor Building.

SUMMARY OF SAFETY EVALUATION:

This modification revises the piping in the Radwaste system to permit flow from P-39 (Condensate/Waste Sludge Mixing Pump) and T-22 (Condensate Drip Tank) to T-26 (Floor Drain Collector Tank) instead of T-24 (Waste Collector Tank) so that T-24 can be taken off line for painting and maintenance while the plant is on line. The new penetration created between the Radwaste Building and the Reactor Building tank room permits service lines to be routed temporarily into the tank room during tank maintenance. The new penetration is not a fire barrier or secondary containment penetration.

70 Core Modifications for Cycle 19 Operations (Mod 97Q230)

DESCRIPTION:

This modification installed 128 fresh GE designed fuel bundles and 4 fresh Durilife 140A control blades. The exposed fuel and control rods were also shuffled to produce the proper core configuration and 6 LPRM strings were replaced with fresh NA-300 LPRMs.

SUMMARY OF SAFETY EVALUATION:

The fuel used conforms to the Loss-of-Coolant Accident analysis, meets the reactivity requirements of the fuel vault and spent fuel pool, and has been accepted by the NRC. License Amendment 100 was issued April 20, 1998.

71 Computer Room Remodeling (Mod 97Q245)

DESCRIPTION:

This modification remodeled the east end of the Plant Administration Building (PAB) Computer Room by walling off a segment of the Computer Room adjacent to the Shift Supervisor (SS) Office and adding a sliding glass transaction window between the new area and the SS Office.

SUMMARY OF SAFETY EVALUATION:

This modification provides an administrative area for use by the Operations Department. There is no direct personnel access to the SS Office from this new area. The swing for an existing emergency exit personnel door into the new area was changed for use as the normal access to and from the new area. A normally locked 3-hour rated fire door is provided in the newly installed wall for maintenance access to a Toshiba cabinet which abuts the new wall inside the Computer Room. Appropriate fire walls are added in areas below false flooring.

72 Redundant Generator Synchronization Permissive Project (Mod 97Q250)

DESCRIPTION:

This modification installs redundant synchronizing check relaying (a sync-check relay and two coupling capacitor voltage transformers) for the 345 KV breakers (8N4 and 8N5) used to connect Monticello to the grid.

SUMMARY OF SAFETY EVALUATION:

The purpose of this modification is to provide another means of preventing the Monticello generator from being synchronized to the NSP transmission system while more than 30° out of phase. This relaying supplements the existing more precise synchronizing protection circuits.

73 RHR Heat Exchanger Temperature Modification Upgrade (Mod 97Q255)

DESCRIPTION:

This modification installed RTDs in the inlets and outlets of the RHR heat exchanger.

SUMMARY OF SAFETY EVALUATION:

The installation of new RHR heat exchanger temperature monitoring equipment will significantly improve the accuracy of the temperatures used in the RHR heat exchanger efficiency test without affecting the reliability of the existing equipment. These changes address concerns raised by the NRC during a 1997 inspection.

74 Containment Penetration Over-pressurization (Mod 98Q010, Part A)

DESCRIPTION:

This part of modification 98Q010 replaces the valve portion of MO-2027 with a double disc gate valve and replaces the thermal overload elements in MCC 143. The modification also installed a rupture disc inside the drywell at the X-18 and 19 penetrations. A gate valve is installed inside the drywell at X-19 for testing.

SUMMARY OF SAFETY EVALUATION:

Modification 98Q010 pertains to a Monticello commitment to address NRC Generic Letter 96-06 relative to the prevention of thermally induced over-pressure in isolated water-filled pipe sections. Part A addresses containment penetrations for the Vessel Head Spray (X-17), Drywell Floor Drain (X-18), and Drywell Equipment Drain (X-19) lines. These permanent pressure-relieving devices do not change the containment penetration isolation scheme described in the USAR.

The gate valve portion of MO-2027 has a hole through the disc on the vessel side. The thermal overload elements for MO-2027 were also replaced to prevent MOV nuisance trips, motor overheating, and improve MOV fuse/thermal overload coordination.

75 Reroute Off-Gas Relief Valve to the Low Pressure (LP) Condenser (Mod 98Q015, Rev. 1)

DESCRIPTION:

This modification reroutes the existing GOLE-107-6" pipe from the High Pressure (HP) condenser penetration #22 to LP condenser penetration #38 (including the addition of 3 supports), and caps the remaining pipe near the 1-1/2" drain that uses the GOLE-107-6" pipe to discharge into the HP condenser. The existing piping will be abandoned in place and can be removed at a later date. A calculation was also performed to determine the adequacy of the baffle on the LP condenser penetration #38.

SUMMARY OF SAFETY EVALUATION:

Events that can result in substantial changes to HP condenser vacuum have the potential to cause the actuation of recombiner steam supply relief valves or off-gas pressure control valves during the transient. Because both of these discharge to a common line that returns to the HP condenser, the flow of steam and non-condensable gases can cause an increase in condenser pressure (reduced vacuum). This reduces the margin to the scram setpoint (as sensed in this condenser) and could affect condensate flow. This modification is intended to address this concern.

76 RHRSW Motor Cooler Flow Improvements (Mod 98Q020)

DESCRIPTION:

This modification upgrades the internal pressure rating of the RHR Service Water pump cooling coil to 100 psig, raises the pressure out of the pressure control valves (PCV-3004 and PCV-3005) to a setting between 35 and 78 psig, and raises the relief setpoint (RV-3038 and RV-3039) to correspond with the change in supply pressure while still preventing over-pressure conditions in the cooling coil. The outlet of the PCV to the confluence of the discharge branches was changed to 1" Schedule 40 pipe.

SUMMARY OF SAFETY EVALUATION:

This scope of this modification includes: 1) evaluation of the cooling flow required for heat removal from the RHRSW pump motor thrust bearing lubricating oil; 2) evaluation of the pressure rating of the associated cooling coils located in the bearing oil reservoir; and 3) makes physical modifications to the system to reduce the piping head loss and increase operating pressure to assure required cooling flow to these cooling coils.

77 Separate Off-Gas Storage & Recombiner Feeders (Mod 98Q025)

DESCRIPTION:

This modification separates the power feeders to the Off Gas Storage building Motor Control Centers (MCCs) from the Off Gas Recombiner MCCs.

SUMMARY OF SAFETY EVALUATION:

This modification improves reliability of the Off Gas Recombiner system by separating the power feeders to the Off Gas Storage building MCCs from the Off Gas Recombiner MCCs. With the new scheme, each MCC (115, 116, 124 and 125) will have a dedicated load center breaker. As typical for other MCC feeder breakers, these breakers are connected to existing load center breaker trip common alarms.

78 Support SS-7 Clamp Modification (Mod 98Q050)

DESCRIPTION:

This modification adds stiffener gussets to the clamp for snubber SS-7A.

SUMMARY OF SAFETY EVALUATION:

This modification addresses a discrepancy in the pipe clamp for snubber SS-7A identified during an In-service Inspection (ISI) exam. A subsequent evaluation determined that snubber SS-7 was attached to the pipe clamp in a typical manner, but snubber SS-7A was attached by two fabricated lugs welded to the bottom half of the same clamp rather than to a separate clamp as shown on the support drawings. The added stiffener gussets have been analyzed to verify that the modified support clamp is in compliance with the original code allowable stress limits.

79 Reactor Level Instrument Line from V15-2"-ED to Drywell Seal Pipe Support Improvements (Mod 98Q055)

DESCRIPTION:

This modification repaired/reconfigured the pipe supports for the 1" Reactor Level Instrument Line from V15-2"-ED to the Drywell Seal.

SUMMARY OF SAFETY EVALUATION:

This modification addresses deficiencies in the pipe supports discovered during reactor vessel re-assembly activities associated with the 1998 refueling outage. Two pipe supports in the reactor cavity were modified. Analysis was then performed using as-built data to assure that the instrument line meets code-allowable stress limits.

EXHIBIT B

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

The following provides a brief description and a summary of changes to commitments established with the NRC by the Monticello Nuclear Generating Plant. These commitments are being identified and reported to the Commission in accordance with guidance provided in the Nuclear Energy Institute (NEI) document "Guideline for the Managing of NRC Commitments," Rev. 2, December 19, 1995. Only one change to commitments was consummated during the interval being considered.

1. Monticello Commitment M81068A

Source Document:	NSP letter dated May 22, 1981 with subject, "Primary Containment Purge & Vent Valve Accumulator Check Valve Leakage Problem" (LER 81-014)
Commitment:	Failed valves have been replaced with new valves and valves will be added to preventative maintenance program.
Change:	This commitment is retracted. When this commitment was made in Licensee Event Report (LER) 81-014 the Primary Containment Purge and Vent Valve T-rings were sealed using an instrument air supply consisting of an accumulator, tubing and an accumulator check valve for each primary containment purge and vent valve. The accumulator check valve was the transition point between the non-safety and safety related portions of the air system. At that time, the function of the accumulator check valves was important for the purge and vent valves to maintain their leak tightness. Per the original commitment, procedure 7260 "Preventative Maintenance for Accumulator Check Valve for P.C. Purge and Vent Valves" was added to the preventive maintenance program. Modification 91Z104 installed a safety grade nitrogen supply to the purge, vent and vacuum breaker T-rings. Because of the addition of a safety grade nitrogen supply, the function of the accumulator check valves is no longer needed and having these check valves in the formal preventative maintenance program is not necessary. Therefore, Procedure 7260 has been deleted.