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Gentlemen:

DOCKETS 50-266 AND 50-301
ANNUAL RESULTS AND DATA REPORT - 1992
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with our February 25, 1993, letter which transmitted unbound copies of the 1992 Annual Results and Data Report, enclosed are 11 bound copies of this report.

We apologize for any inconvenience this may have caused.

Sincerely,

Bob Link
Vice President
Nuclear Power

DAW/jg

Enclosures

cc: NRC Resident Inspector
NRC Regional Administrator, Region III

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WISCONSIN ELECTRIC

POWER COMPANY

POINT BEACH NUCLEAR PLANT

UNIT NOS. 1 AND 2

ANNUAL RESULTS AND

DATA REPORT

1992

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110/18

U.S. Nuclear Regulatory Commission
Docket Nos. 50-266 and 50-301
Facility Operating License Nos.
DPR-24 and DPR-27

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DPR-24 and DPR-27

PREFACE

This Annual Results & Data Report for 1992 is submitted in accordance with Point Beach Nuclear Plant, Unit Nos. 1 and 2, Technical Specification 15.6.9.1.B and filed under Docket Nos. 50-266 and 50-301 for Facility Operating License Nos. DPR-24 and DPR-27, respectively.

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I. INTRODUCTION

The Point Beach Nuclear Plant, Units 1 and 2, utilize identical pressurized water reactors rated at 1518 MWt. Each turbine-generator is capable of producing 497 MWe net (524 MWe gross) of electrical power. The plant is located ten miles north of Two Rivers, Wisconsin, on the west shore of Lake Michigan.

II. HIGHLIGHTS

UNIT 1

Highlights for the period January 1, 1992, through December 31, 1992, included a 55-day refueling/maintenance outage. Major work items included ultrasonic testing of fuel rods, testing and reconditioning of the main steam isolation valves, and as-built walkdowns of the reactor protection and safeguards systems. The unit experienced an ESF actuation caused by an inadvertent deenergization of a 4160 volt safeguards bus; a precautionary load reduction to facilitate live transfer of DC distribution cabinets D11 and D12 from the normal DC bus D01 to a temporary DC bus for installation of a new D01 bus; the generator was taken off-line for less than an hour to conduct turbine overspeed tests; and the unit was taken off-line to repair a heater drain tank manway leak.

Unit 1 operated at an average capacity factor of 84.7% (MDC net) and an electrical/thermal efficiency of 32.5%. The unit and reactor availability were 84.4% and 84.6%, respectively. Unit 1 generated its 72 billionth kilowatt hour on March 13, 1992; its 73 billionth kilowatt hour on July 31, 1992; and its 74 billionth kilowatt hour on October 21, 1992.

UNIT 2

Highlights for the period January 1, 1992, through December 31, 1992, included a 53-day refueling/maintenance outage. Major work items included containment integrated leak rate testing; eddy current inspection of steam generators; D02 battery replacement; and mechanical and electrical modifications on the main steam isolation valves. The unit experienced reduced power for a few weeks because startup reactor coolant system precision flow measurements were discovered to be very close to Technical Specification limits; and a scheduled shutdown for mid-cycle testing of the main steam isolation valves.

Unit 2 operated at an average capacity factor of 86.2% (MDC net) and an electric/thermal efficiency of 32.8%. The unit and reactor availability were 85.3% and 85.5%, respectively. Unit 2 generated its 72 billionth kilowatt hour on March 8, 1992; its 73 billionth kilowatt hour on May 28, 1992; its 74 billionth kilowatt hour on August 17, 1992; and its 75 billionth kilowatt hour on December 31, 1992.

III. Amendments to Facility Operating Licenses

During 1992, there were four amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1 and four amendments issued to Facility Operating License DPR-27 for Point Beach Nuclear Plant Unit 2. The license amendments are listed by date of issue and summarized below:

Amendment 130 to DPR-24, Amendment 134 to DPR-27, April 8, 1992: The amendments changed the minimum volume of water available in the condensate storage tanks from 10,000 to 13,000 gallons per unit. The TS Basis for this change was also revised.

Amendment 133 to DPR-24, Amendment 137 to DPR-27, July 31, 1992: The amendments require each auxiliary feedwater (AFW) pump to be started on a quarterly basis. The TS Basis for this change was also revised. The amendments also clarified auxiliary feedwater pump out-of-service limitations.

Amendment 134 to DPR-24, Amendment 138 to DPR-27,September 18, 1992: The amendments add limiting conditions of operation to prescribe when the 480 volt safeguards buses may be tied; when the 4160 volt safeguards buses may be tied; and when these safeguards buses must be powered from their normal supplies. The TS Basis for this change was also revised.

Amendment 135 to DPR-24, Amendment 139 to DPR-27, November 3, 1992: The amendments authorize the annual diesel generator inspection to be performed at an 18-month interval for the current inspection.

Amendment 136 to DPR-24, Amendment 140 to DPR-27, November 23, 1992: The amendments incorporate a fifth station battery and provide testing requirements for safety-related station batteries.

IV. 10 CFR 50.59

PROCEDURE CHANGES

1. ABB Impell, PI-0087-00023-1, -2, -3, (Minor), Cable Verification Program, Revision 0. (New Procedure) This project was initiated to statistically establish an overall 95% confidence level of the cable routing recorded in the cable and raceway data system (CARDS) computer program. The routing of a representative random sample of the safety-related cables is verified by walkdowns. Some of the cable walkdowns are visual only; some aided by an electronic cable tracing device. No leads are lifted, no circuits modified and no circuits made inoperable. The tracing device injects a low energy signal on the cable to be traced. The cable is then traced using the receiving unit to monitor the injected signal.

   Summary of Safety Evaluation: The cable tracing equipment introduces a low level (less than 100 mV) frequency (7 Khz to 455 Khz) signal on to the cable to be traced. This signal does not affect relays, circuit breakers, or instrumentation. The circuits to be traced are not modified. The test equipment does not cause any bistables to change state or any control functions to actuate. Capacitive coupling of the test signal into adjacent cables does not cause a problem since cables of redundant circuits are separated from the circuit traced. Also, the capacitive coupled signal into an adjacent cable does not affect the cable's function since the signal being induced into the cables is less than the original test signal.

The cable tracing is performed by a combination of visual inspection and test equipment implementation, and does not affect prior technical commitments to the NRC regarding the design, operation or function of any safety-related SSC described in the PBNP FSAR. (SER 92-087)

2. CAMP-009, (Minor), Chlorination/Dechlorination System Operation, Revision 5, dated April 23, 1992. (Permanent)

This revision allows SW-2869 or SW-2870 to be shut while performing prolonged chlorination of an individual unit's service water system. SW-2869 and SW-2870 are the motor operated valves (MOVs) which provide isolation from the north and south service water headers to the west service water header.

Summary of Safety Evaluation: During normal operation, shutting one west service water header isolation MOV has no impact on system operation since service water is supplied to the west header components by the header whose isolation MOV to the west header is open.

Having one west service water header isolation MOV shut does not impact system operation in the event of a large rupture of any service water header. This malfunction is addressed in Section 6.3 of AOP-9A, "Service Water System Malfunction." The immediate action is to shut all four service water isolation MOVs. Since one of these MOVs already is shut, there is no impact on the performance of AOP-9A.

Having one west service water header isolation MOV shut could have an impact in the event of a blockage in a main header. Such a blockage in the proper location could stop service water flow to as many as six containment fan coolers and two component cooling heat exchangers. Data collected during the performance of IT-07, "Service Water Pumps and Valves," has not shown evidence of such a blockage. The probability of such a blockage rapidly forming is extremely small due to the large size of the pipe and the high flow rate through the main headers. In the event of a blockage, the isolation MOV which is shut would be opened to restore flow to the affected components. Since the isolation MOVs have handwheels, they can be opened even in the event of a loss of electrical power. Flow could also be restored by opening the service water cross connect valves at the component cooling heat exchangers. (SER 92-040)

3. CP-212, (Minor), SAS/PPCS Computers: Software Initialization, Revision 4, dated March 3, 1992. (Permanent)

Summary of Safety Evaluation: This revision adds constants K0031 and K0320 to Appendix A. The constants were added because corrections were made to the calculation of reactor thermal output (RTO) on the PPCS. The conversion factor for gpm to kft^3/hr in the charging line and letdown line heat rate calculation used in the calculation of RTO was incorrect. This was corrected to the appropriate addressable point constant, K0031. The change increases indicated (calculated) RTO by about 1.3 MWt at normal full power conditions. The use of PPCS temperature point M126 for determining the specific volume in the charging line heat rate calculation used in the calculation of the RTO was incorrect. This was changed to a more correct yet conservative value by using addressable point constant K0320. This change decreases indicated (calculated) RTO by about 0.3 MWt at normal full power conditions.

Because the described changes serve to more correctly calculate RTO, setpoints that are directly or indirectly based on the calculated RTO are determined more correctly. Therefore, there are no adverse safety concerns regarding setpoints and these changes. (SER 92-012)

4. CSP-C.2, (Major), Response to Degraded Core Cooling, Revision 7, dated February 4, 1992. (Permanent)

Summary of Safety Evaluation: The revision changes the wide range reactor vessel level indication as a result of MR 88-018*K and requires new EOP setpoint values.

All of the new values are more conservative numbers in that required actions are taken at higher indicated levels. Only the more conservative adverse containment values are currently used. This revision adds normal containment value, and the adverse containment value is indicated by brackets as in other EOP setpoints. The narrow range reactor vessel level values do not change. The normal and adverse values for narrow range are the same value, so this value is only listed once and is not changed. (SER 88-095-06)

5. CSP-C.3, (Major), Response to Saturated Core Cooling, Revision 5, dated February 4, 1992. (Permanent)

Summary of Safety Evaluation: Wide reactor vessel level was modified by MR 88-018*K. The values provided are the more conservative adverse containment setpoints. The normal containment setpoints are added and the adverse containment values are enclosed in brackets as for other setpoints. Narrow range values are not affected and adverse containment values are not necessary since the calculations show that the normal and adverse values are the same. (SER 89-099-04)

6. CSP-S.1, (Major), Response to Nuclear Power Generation/ATWS, Revision 6, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: The revision changed Step 12 to include verification of the main steam isolation bypass valve shut. This item was not previously checked because the valve is normally shut during full power operation. As a result of LER 266/91-015-00, the main steam isolation bypass valve is verified shut to ensure steam generator isolation. The steam generator isolation list also added the P29 auxiliary feedwater pump/radwaste steam isolation valve. This isolates the P29 governor sensing line and the radwaste steam supply if aligned to the affected steam generator. (SER 89-081-05)

7. CSP-S.1, (Major), Response to Nuclear Power Generation/ATWS, Revision 8, dated September 30, 1992. (Permanent)

CSP-S.1 was revised to include nuclear instruments used during steps which require reading them. This is necessary to ensure the operators include the use of N40 since it is the only environmentally qualified nuclear instrument.

Summary of Safety Evaluation: Regulatory Guide 1.97 requires the use of the qualified instrument during accident conditions for a parameter that has both a qualified instrument and a non-qualified instrument available. The use of N40 improves the safe operation of the plant as it is environmentally qualified and provides a redundant method for evaluating core reactivity. (SER 89-081-06)

8. ECA 1.1, (Major), Loss of Containment Sump Recirculation, Revision 8, dated February 4, 1992. (Permanent)

Summary of Safety Evaluation: The revision results in securing the safety injection (SI) pump, securing the suction to the boric acid storage tank (BAST), placing the suction to the refueling water storage tank (RWST) on service and starting the SI pump. This operation is within SI system design and does not present an unreviewed safety

question. TS 15.3.2 does not apply under the conditions that would cause this procedure to be entered.

Solidification of the higher concentration boric acid solution in the SI pumps and piping is prevented by procedural controls to minimize the time the SI pumps are off. The SI pumps were stopped and restarted to perform flushing prior to EOP development (pre-TMI) and no solidification problems were experienced. (SER 88-093-07)

9. ECA 2.1, (Major), Uncontrolled Depressurization of Both Steam Generator, Revision 12, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: The revision changed Step 1 to include verification of the main steam isolation bypass valve shut. This item was not previously checked because the valve is normally shut during full power operation. As a result of LER 266/91-015-00, the main steam isolation bypass valve is verified shut to ensure steam generator isolation. The steam generator isolation list also added the P29 auxiliary feedwater pump/radwaste steam isolation valve. This isolates the P29 governor sensing line and the radwaste steam supply if aligned to the affected steam generator.

The method of switching suction from the boric acid storage tank (BAST) to the refueling waster storage tank (RWST) during the safety injection (SI) flush was changed. This change was necessary since the previous method of aligning for an SI flush resulted in transferring water inventory from the RWST to the BAST due to the height differences of both tanks, resulting in diluting the BAST.

Steps 14 and 13 were reversed to ensure diesel status is checked prior to starting an instrument air compressor. (SER 89-033-12)

10. ECA-3.2, (Major), SGTR with Loss of Reactor Coolant - Saturated Recovery Desired, Revision 10, dated February 4, 1992. (Permanent)

Summary of Safety Evaluation: The RVLIS differential pressure detectors are located inside containment where adverse containment conditions may affect the indicated level. This is different from the Westinghouse Owner's Group Emergency Response Guideline (ERG) design where the detectors are located outside containment. The adverse containment value for the wide range reactor vessel level was added to Steps 13, 17, and the foldout page. Not all reactor vessel level setpoints are affected since under some conditions, the normal and adverse containment setpoints are the same value. Modification 88-018*K changes wide range RVLIS setpoints [F.14] and F.6 which are now [60'] and 50' respectively. (SER 89-034-10)

11. ECA-3.3, (Major), SGTR Without Pressurizer Pressure Control, Revision 8, dated February 4, 1992. (Permanent)

Summary of Safety Evaluation: Wide range reactor vessel level indication was deleted to be consistent with the Westinghouse Owner's Group Emergency Response Guidelines (ERG). The setpoint used was not consistent with the Westinghouse setpoint description and was not conservative. This substep was added in Revision 3 of ECA-3.3 to provide uniformity to similar steps. (SER 89-035-08)

12. EOP-1, (Major), Loss of Reactor or Secondary Coolant, Revision 12, dated February 27, 1992. (Permanent)

Summary of Safety Evaluation: The first note for Step 11 was changed to ensure the diesel is properly loaded. Guidance in Appendix A was included to establish when the

continuous diesel loading limit may be exceeded. Steps 10 and 11 were reversed to ensure diesel loading is verified prior to starting an instrument air compressor on a diesel that may be heavily loaded.

The changes to Step 20 reflect the change in the normal operating position of the residual heat removal (RHR) pump room drain valves from open to shut. Valves WL-4100 and WL-4101 are shut so the individual RHR pump room level alarms indicate which RHR pump seal is failing.

EOP Setpoint L.3 was recalculated to reflect a new flow transmitter in train "B" RHR flow as required by MR 88-018*D. Setpoint L.3 was changed from 350 gpm to 275 gpm and Setpoint L.13 was added. (SER 88-087-12)

13. EOP 1.1, (Major), SI Termination, Revision 10, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: The revision results in securing the safety injection (SI) pumps and then shifting the suction to the refueling water storage tank (RWST). This change is within the scope of the intent of the SI system and does not pose an unreviewed safety question.

TS 15.3.2, which addresses boric acid heat tracing preventing crystallization, does not apply under the conditions that would cause this procedure to be entered. The concern of crystallization is addressed by limiting the time the SI lines are in a no-flow condition. The reversed method of switching suction from the boric acid storage tank (BAST) to the RWST was used prior to existence of the EOP procedures (pre-TMI) without indication of boric acid crystallization. (SER 88-088-10)

14. EOP 1.3, (Major), Transfer to Containment Sump Recirculation, Revision 9, dated October 11, 1991. (Temporary)

Summary of Safety Evaluation: Normal operating position of the residual heat removal (RHR) pump room drain valves was changed from open to shut. Valves WL-4100 and WL-4101 are shut so the individual RHR pump room level alarms indicate which RHR pump seal is failing. (SER 88-089-11)

15. EOP 1.3, (Major), Transfer to Containment Sump Recirculation, Revision 10, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: The normal operating position of the residual heat removal (RHR) pump room drain valves was changed from open to shut. Valves WL-4100 and WL-4101 are shut so the individual RHR pump room level alarms indicate which RHR pump seal is failing.

EOP Setpoint L.7 was recalculated as a result of a new flow transmitter for train "B" RHR flow per MR 88-018*D. Setpoint L.7 changed from 350 gpm to 1100 gpm.

Step 14i was repositioned after Step 14l to ensure P10A RHR pump suction from the refueling water storage tank (RWST), (SI-856A) is shut before opening the suction from containment sump B, (SI-851A). This helps to prevent possible contamination of the RWST by both valves being open during an idle pump condition. (SER 88-089-10)

16. EOP-1.4, (Major), Transfer to Containment Sump Recirculation, One Train Inoperable, Revision 5, dated October 11, 1991. (Temporary)

Summary of Safety Evaluation: The normal operating position of the residual heat removal (RHR) pump drain to sump control valves WL-4100 and WL-4101 was changed

from shut to open. This allows the individual RHR pump room alarms to be used for determination of location of RHR pump seal leakage. (SER 89-100-11)

17. EOP-1.4, (Major), Transfer to Containment Sump Recirculation, One Train Inoperable, Revision 6, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: The procedure was revised to reflect replacement of "B" train residual heat removal (RHR) discharge flow transmitter per MR 88-018*D. EOP Setpoint L.10 (400 gpm) changed to 450 gpm. The basis for EOP setpoint L.10 is to prevent RHR pump runout when supplying the suction of the safety injection (SI) pump. The normal operating position of the RHR pump drain to sump control valves WL-4100 and WL-4104 was changed from shut to open. This allows the individual RHR pump room alarms to be used for determination of location of RHR pump seal leakage. (SER 89-100-10)

18. EOP-2, (Major), Faulted Steam Generator Isolation, Revision 6, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: Verification of main steam isolation bypass valve shut was added. The item was not previously checked because the valve is normally shut during full power operation. As a result of LER 266/91-015-00, the main steam isolation bypass valve is verified shut to ensure steam generator isolation. The steam generator isolation list also added the P29 auxiliary feedwater pump/radwaste steam isolation valve. This isolates the P29 governor sensing line and the radwaste steam supply if aligned to the affected steam generator.

Steam generator isolation substeps were duplicated to prevent selecting the incorrect valve. A substep was added to skip isolation substeps for the unaffected steam generator. (SER 89-038-05)

19. EOP-3, (Major), Steam Generator Tube Rupture, Revision 13, dated March 25, 1992. (Permanent)

Summary of Safety Evaluation: Verification of main steam isolation bypass valve shut was added. This item was not previously checked because the valve is normally shut during full power operation. As a result of LER 266/91-015-00, the main steam isolation bypass valve is verified shut to ensure steam generator isolation. The steam generator isolation list also added the P29 auxiliary feedwater pump/radwaste steam isolation valve. This isolates the P29 governor sensing line and the radwaste steam supply if aligned to the affected steam generator.

Steam generator isolation substeps were duplicated to prevent selecting the incorrect valve. A substep was added to skip isolation substeps for the unaffected steam generator.

The method of switching suction from the boric acid storage tank (BAST) to the refueling water storage tank (RWST) during the safety injection (SI) flush was changed. The previous method of aligning for an SI flush resulted in transferring water inventory from the RWST to the BAST due to the height differences of both tanks, resulting in diluting the BAST. The changes which result in securing the SI pumps and then shifting the suction to the RWST are within the scope of the intent of the SI system.

TS 15.3.2, which addresses boric acid heat tracing to prevent crystallization, does not apply under the conditions that would cause this procedure to be entered. The concern of the boric acid crystallizing is addressed by limiting the time the SI lines are in a no-flow condition. The method of switching suction from the BAST to the RWST was

used prior to existence of the EOPs (pre-TMI) without an indication of boric acid crystallization. (SER 88-090-12)

20. HP 3.2.5. (Minor). Radioactive Material Storage in Warehouse 2, Revision 4, dated March 9, 1992. (Permanent)

Summary of Safety Evaluation: This evaluation addresses the long-term storage of radioactive materials in Warehouse 2. Materials which are stored in this area consist primarily of tools and spare parts which are single use or periodic use with long periods of non-use. The materials contain low-level fixed contamination, but no loose contamination which may pose an airborne consideration. The site is not used for radioactive waste storage.

The storage of radioactive material in Warehouse 2 is in accordance with the licensing bases documents, and does not pose an unreviewed safety question, based on the completion and adherence to the following items: Health Physics must impose controls to limit access to the storage area to those persons authorized for entry by plant supervisors and health physics personnel; the area of Warehouse 2 where the radioactive materials are to be stored shall be surveyed, classified, and conspicuously posted with a sign or signs bearing the radiation caution symbol and the words: CAUTION (or DANGER) - Radioactive Materials; all radioactive materials shall be stored off the ground on pallets or shelving; no radioactive waste is to be stored in Warehouse 2; the storage area must be secured so as to restrict unauthorized removal of the (licensed) materials; and amend the PBNP fire attack plan to address the potential of contaminated fire suppression system runoff and for potential of contamination of fire-fighting equipment if a fire should ever start in Warehouse 2, followed by fire suppression system activation. The fire attack plan addresses radioactive materials being stored in this warehouse. (SER 92-011)

21. HP 9.4. (Minor). Temporary Lead Shielding on Letdown Line, Revision 5, dated June 17, 1991. (Permanent)

Summary of Safety Evaluation: The revision adds two layers of temporary shielding on the 2" letdown line just up and down stream of motor-operated valve 1LCV-427 for ALARA purposes. Plant conditions are limited to cold or refueling shutdown.

The letdown line remains seismic because plant conditions limit pressure and thermal stresses; extremely short spans for whip supports limit potential displacements and resonances; increase in stress to the seismic vertical component is ≈12% versus a large margin between calculated values and Code allowables; axial loads are minimal due to the nature of attachment of temporary lead shielding; valves are blocked to limit deflection; and the configuration was field-verified.

The temporary modification has no impact on system function or capabilities for the condition identified. There is no impact on the cavity draindown analysis. (SER 88-102-01)

22. HPIP 4.59. (Minor). Operation of Portucount-Plus Respirator Fit Tester, Revision 0, dated September 8, 1992. (New Procedure)

Summary of Safety Evaluation: The FSAR states, "Fitting procedures consist of placing the mask on the face and tightening the straps until the individual believes that he has a good fit. A respirator test booth or smoke test is used to ensure the scalability of the face masks." The existing method for fit testing consists of using a booth. However, the Portucount-Plus uses ambient air instead of a "booth" and compares the dust concentrations in the air to those concentrations inside of a mask worn by an individual.

The intent and implementation of the FSAR description of a respirator fit test are identical to the Portucount system. The FSAR description changed to "...A respirator test booth/system or smoke test is used..." to accommodate use of both a "booth" type system which may be used as a backup, and the Portucount. (SER 92-078)

23. 1ICP-02.003A, Unit 1, (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.003A-1, Unit 1, (Minor), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.003A-2, Unit 1, (Minor), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.003B, Unit 1, (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.003B-1, Unit 1, (Minor), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.003B-2, Unit 1, (Minor), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003A, Unit 2, (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003A-1, Unit 2, (Minor), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003A-2, Unit 2, (Minor), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003B, Unit 2, (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003B-1, Unit 2, (Minor), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.003B-2, Unit 2, (Minor), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review including Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. The new procedures were created to be unit and train specific. A new minor procedure was created to perform testing when a unit is at less than 50% reactor power. Steps were added listing conditions necessary to verify the equipment has been returned to service. Annunciator status is now checked during the test as opposed to after a complete train is tested. Acceptance criteria were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-022)

24. 1ICP-02.003A (Unit 1), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 1, dated July 10, 1992. (Permanent)

1ICP-02.003A-1 (Unit 1), (Minor), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 2, dated July 10, 1992. (Permanent)

1ICP-02.003B (Unit 1), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 1, dated July 10, 1992. (Permanent)

1ICP-02.003B-1 (Unit 1), (Minor), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 2, dated July 10, 1992. (Permanent)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. The Limiting Conditions for Operation were changed. Only one reactor protection system (RPS) train is tested at a time and the redundant RPS train must be operable for performance. Dropped rod mode switch position nomenclature was changed from BYPASS to BYPASS/TEST as a result of MR 91-226. Steps referring to turbine runback were also deleted as a result of MR 91-266. The step to release train shunt block pushbutton was moved to facilitate ease of performance in and when checking the auxiliary feed pump start circuit using test light. After matrix testing is complete, the step to close RPS logic test panel door was moved to after bypass breaker is fully racked out to ensure bypass breaker is tripped. Control room panel 1C04 "PRZR HI PRESS" status lights were changed to show correct channel status lights. (SER 92-022-01)

25. 2ICP-02.003A (Unit 2), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 1, dated July 10, 1992. (Permanent)

2ICP-02.003A-1 (Unit 2), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 2, dated July 10, 1992. (Permanent)

2ICP-02.003B (Unit 2), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 1, dated July 10, 1992. (Permanent)

2ICP-02.003B-1 (Unit 2), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 1, dated July 10, 1992. (Permanent)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. The Limiting Conditions for Operation were changed. Only one reactor protection system (RPS) train is tested at a time and the redundant RPS train must be operable for performance. The step to release train shunt block pushbutton was moved to facilitate ease of performance. Individual checkoff lines were added to each trip combination for the turbine auto stop matrix and when checking the auxiliary feed pump start circuit using test light at RPS test. After matrix testing is complete, the step to close RPS logic test panel door was moved to after the bypass breaker is fully racked out to ensure the bypass breaker is tripped. Control room panel 2C04 "PRZR HI PRESS" status lights were changed to show correct channel status lights. (SER 92-022-02)

26. 1ICP-02.007 (Unit 1), (Major), Nuclear Instrumentation Power Range Channels Monthly Surveillance Test, Revision 0, May 28, 1992. (New Procedure)

1ICP-02.007-1 (Unit 1), (Minor), Nuclear Instrumentation Power Range Channels Monthly Surveillance Test, Revision 0, May 28, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, MR 91-226, associated drawings, and CHAMPS with format upgrade and human factoring. As a result of MR 91-226, the turbine runback feature from a dropped rod was removed. The NIS dropped rod mode switch position was relabeled from BYPASS to BYPASS/TEST position. Steps were added listing conditions

necessary to verify the equipment has been returned to service. Acceptance criteria statements were added to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-053)

27. 2ICP-02.007 (Unit 2), (Major), Nuclear Instrumentation Power Range Channels Monthly Surveillance Test, Revision 0, dated May 28, 1992. (New Procedure)

2ICP-02.007-1 (Unit 2), (Minor), Nuclear Instrumentation Power Range Channels Monthly Surveillance Test, Revision 0, dated May 28, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. Steps were added listing conditions necessary to verify the equipment has been returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-054)

28. 1ICP-02.011 (Unit 1), (Major), Analog Rod Position Monthly Surveillance Test, Revision 0, dated July 6, 1992. (New Procedure)

1ICP-02.011-1 (Unit 1), (Major), Analog Rod Position Monthly Surveillance Test, Revision 0, dated July 6, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, MR 91-226, associated drawings, and CHAMPS with format upgrade and human factoring. As a result of MR 91-226, the turbine runback feature from a dropped rod was removed. The turbine runback/rod stop defeat switch and circuitry and associated "Turbine Runback Defeated" annunciator was disabled. Procedure steps associated with these functions were deleted. Rod position indication (RPI) system test panel indicating light nomenclature was changed. Steps associated with the indicating light were revised. Steps were added listing conditions necessary to verify the equipment has been returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-058)

29. 2ICP-02.011 (Unit 2), (Major), Analog Rod Position Monthly Surveillance Test, Revision 0, dated July 6, 1992. (New Procedure)

2ICP-02.011-1 (Unit 2), (Major), Analog Rod Position Monthly Surveillance Test, Revision 0, dated July 6, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. Steps were added listing conditions necessary to verify the equipment was returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-059)

30. 1ICP-02.013, Unit 1, (Major), 4.16 KV Undervoltage 4 Matrix Relays Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

1ICP-02.013-1, Unit 1, (Minor), 4.16 KV Undervoltage 4 Matrix Relays Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.013, Unit 2, (Major), 4.16 KV Undervoltage 4 Matrix Relays Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

2ICP-02.013-1, Unit 2, (Minor), 4.16 KV Undervoltage 4 Matrix Relays Monthly Surveillance Test, Revision 0, dated March 20, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. Limiting Conditions for Operation (LCO) statements were included. The procedures do not place the plant in an LCO. (SER 92-023)

31. 1ICP-02.021 (Unit 1), (Minor), Analog Rod Position Shutdown Surveillance Test, Revision 0, dated May 8, 1992. (New Procedure)

1ICP-02.021-1 (Unit 1), (Minor), Analog Rod Position Shutdown Surveillance Test, Revision 0, dated May 8, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, MR 91-226, associated drawings, and CHAMPS with format upgrade and human factoring. As a result of MR 91-266, the turbine runback feature from a dropped rod were removed and steps associated with these functions were deleted. Rod position indication (RPI) system test panel indicating light nomenclature was changed. This indicating light is still active. Steps were added listing conditions necessary to verify the equipment is returned to service. Acceptance criteria statements have been included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-044)

32. 2ICP-02.021 (Unit 2), (Major), Analog Rod Position Shutdown Surveillance Test, Revision 0, dated May 28, 1992. (New Procedure)

2ICP-02.021-1 (Unit 2), (Minor), Analog Rod Position Shutdown Surveillance Test, Revision 0, dated May 28, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. Steps were added listing conditions necessary to verify the equipment is returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-052)

33. 1ICP-02.022 (Unit 1), (Major), Nuclear Instrumentation System Power Range Channels Shutdown Surveillance Test, Revision 0, dated December 17, 1992. (New Procedure)

1ICP-02.022-1 (Unit 1), (Minor), Nuclear Instrumentation System Power Range Channels Shutdown Surveillance Test, Revision 0, dated December 17, 1992. (New Procedure)

Summary of Safety Evaluation: A technical review included Technical Specifications, FSAR, MR 91-226, associated drawings, and CHAMPS with format upgrade and human factoring. Correct nomenclature was included to reflect changes made by MR 91-226. Steps were added listing conditions necessary to verify the equipment has been returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 93-001)

34. 2ICP-02.022 (Unit 2), (Major), Nuclear Instrumentation System Power Range Channels Shutdown Surveillance Test, Revision 0. (New Procedure)

2ICP-02.022-1 (Unit 2), (Minor), Nuclear Instrumentation System Power Range Channels Shutdown Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review included Technical Specifications, FSAR, MR 91-227, associated drawings, and CHAMPS with format upgrade and human factoring. Steps were added to the procedure listing conditions necessary to verify the equipment was returned to service. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-101)

35. 1ICP-02.023A (Unit 1), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

1ICP-02.023A-1 (Unit 1), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

1ICP-02.023B (Unit 1), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

1ICP-02.023B-1 (Unit 1), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

2ICP-02.023A (Unit 2), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

2ICP-02.023A-1 (Unit 2), (Major), Reactor Protection System Logic Train A Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

2ICP-02.023B (Unit 2), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

2ICP-02.023B-1 (Unit 2), (Major), Reactor Protection System Logic Train B Monthly Surveillance Test, Revision 0, dated July 16, 1992. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, associated drawings, and CHAMPS with format upgrade and human factoring. The procedures perform testing when the unit is <50% reactor power. Steps were included listing conditions necessary to verify the equipment was returned to service. Annunciator status is checked during testing as opposed to after a complete train is tested. Acceptance criteria statements were included to describe conditions and results necessary to satisfy Technical Specification requirements. (SER 92-060)

36. ICP 11.454, (Minor), I&C Surveillance Test Safeguards System Logic - Post As-Built, Revision 0, dated May 12, 1992. (New Procedure)

Summary of Safety Evaluation: This test is performed prior to leaving cold shutdown to verify ESF circuitry. ORT 3 is performed after this test to verify ESF output circuitry.

Testing of the logic relay is normally done using ICP 2.19; however, computer, status, and annunciator inputs are not verified during this test. Analog signal calibration is normally done using ICP 2.20 on a channel by channel basis.

Analog inputs for steam line pressures and flows, loop temperatures, pressurizer pressure, and steam generator levels are simulated to reset bistable elements and logic relays. The testing requires simulating the inputs of the same sense value on different channels simultaneously to allow complete logic testing.

Unblocking and blocking of SI is performed in conjunction with ICP 10.1 (Safeguards' and AMSAC System Bypass) to ensure an ESF signal is not generated. Completion of the test performs relay continuity checks to verify output relay integrity prior to ORT 3.

During the continuity checks, similar steps are taken to ensure that an ESF signal is not generated. (SER 92-033-01)

37. ICP 11.455, (Minor), Retest of Reactor Protection System Following As-Built Wire Tracing, Revision 0, dated May 7, 1992. (New Procedure)

This test directs performance of modified ICP 2.17, ICP 2.15, and ICP 2.9, as a retest for reactor protection system as-built wire tracing.

Summary of Safety Evaluation: The safety function of the reactor protection system is to open reactor trip and bypass breakers when plant parameters are sensed out of the normal condition. Since this test uses red tags to prevent RCCAs from being energized, the safety function of the reactor protection system is met.

The only significant deviation from existing ICP 2.17, 2.15, and 2.9, involves testing the source range high volts cutoff interlock. This deenergizes both source range channels for a short term. Plant conditions specified for this test are refueling operations in progress, and the wide-range NI channel be operable. Both wide-range NI channels are expected to be deenergized for a period of about 1 minute.

Use of jumpers as directed in ICP 11.455 does not constitute a temporary modification, since the reactor protection racks are not in service during the time the procedure is performed. (SER 92-045)

38. 2ICP 11.465, Unit 2, (Minor), Parent Procedure for Westinghouse NSID-EIS-85-11, and NSID-EIS-85-11, Full Length Rod Control System Maintenance, Revision 0, dated May 7, 1992. (New Procedure)

Summary of Safety Evaluation: Performance of Westinghouse NSID-EIS-85-11, requires reactor trip breakers be shut and a rod drive MG be energized for a portion of the maintenance. The maintenance includes attaching a "dummy" CRDM coil set to the rod control cabinets, simulating shimming rods, and obtaining current traces and other measurements on the rod control system.

Because the unit is shutdown, the only accident of concern is inadvertent RCCA withdrawal from a subcritical condition. Reactor protection system response to this accident is automatic reactor trip on source, intermediate, or power range high neutron flux. Since the rod control cabinet output fuses to the CRDM coils are red tag removed so the CRDM coils are ensured to remain deenergized. Plant conditions which require reactor trip breakers to be open are considered. The condition is red tagging rod control cabinet 1AC, 2AC, and 1BD output fuses to CRDMs removed. It is also understood that installation of "dummy" fuses in select locations of rod drive cabinet output circuitry is in keeping with the intent of maintaining CRDM coils deenergized. (SER 92-091)

39. 2ICP 11.466, Unit 2, (Minor), As-Built Wire Tracing of the Reactor Protection Relay Racks, Revision 0, dated September 25, 1992. (New Procedure)

2ICP 11.468, Unit 2, (Minor), As-Built Wire Tracing of the Safeguards Relay Racks, Revision 0, dated September 25, 1992. (New Procedure)

The safeguards and reactor protection as-built project involves hand-over-hand wire tracing in the respective protection and safeguards racks. To reduce the chance for electric shock, reactor protection control power and safeguards control power is deenergized when wire tracing is performed. Both trains of reactor protection control power is deenergized simultaneously, and one train of safeguards control power is deenergized during performance of the wire tracing.

Summary of Safety Evaluation: If two trains of reactor protection are operable, the unit must be maintained in at least hot shutdown. For the wire tracing activities, both trains of RPS are simultaneously deenergized. To ensure no positive reactivity is added from control rods, control rod drive mechanisms are deenergized. One of the following sets of breakers is tagged open or racked out: reactor trip and bypass breakers; or rod drive MG supply breakers; or rod drive MG load breakers.

If two trains of safeguards are not operable, the unit must be in cold shutdown. When the plant is in cold shutdown, no safeguards system operability is required. Only one train of safeguards control power will be deenergized at a time. When a train of control power is deenergized, automatic actions and interlocks associated with safeguards do not function. However, manual operation of either train is not affected. Thus, neither train's availability for normal shutdown operation, such as decay heat removal, is affected by deenergizing its safeguards control power.

Containment ventilation isolation (CVI) is required to be operable during refueling operations. Operable means that automatic CVI occurs when high containment radiation is sensed. If CVI is not operable, then the CVI valves must be shut. When one train of safeguards control power is deenergized, the corresponding train CVI valves do not receive an automatic closure signal on high containment radiation. Manual operation of the CVI valves from their switches in the control room is not affected. Administrative controls ensure that the CVI function occurs if required. (SER 92-081)

40. 2ICP 11.467, Unit 2, (Minor), Retest of Reactor Protection System Following As-Built Wire Tracing, Revision 0, dated October 22, 1992. (New Procedure)

Summary of Safety Evaluation: This procedure directs performance of modified ICP 2.17, ICP 2.15, and ICP 2.9 as a retest for reactor protection system (RPS) as-built wire tracing. The testing simulates reactor plant parameters necessary to clear reactor trips. The parameters include steam generator narrow range level, pressurizer pressure, pressurizer level, loop flows, condenser vacuum to RPS, and reactor coolant and circulating water pumps breaker position to RPS. Testing of trip, bypass, and interlock functions is performed per temporary changes made to ICP 2.17, ICP 2.15 and ICP 2.9. These procedures check indications not annually checked as part of logic testing, such as annunciator, computer inputs, and status lights for all signals, verification of parallel contacts, and parallel relays. Additionally, the source range deenergization interlock is checked by momentarily deenergizing both SRNI channels.

The safety function of the reactor protection system is to open reactor trip and bypass breakers when plant parameters are sensed out of the normal condition. Since this procedure uses red tags to prevent RCCAs from being energized, the safety function of the reactor protection system is met.

Operators are used to signify on instrument meter faces which instruments have simulated parameters. Other parameters may be simulated if required to clear reactor trips, which were not anticipated to be in the tripped state for the specified plant conditions.

The only significant deviation from existing ICP 2.17, 2.15 and 2.9, involves testing the source range high volts cutoff interlock. This deenergizes both source range channels for a short time. Plant conditions specified are that no refueling operations in progress, and the wide-range NI channel be operable. Both wide-range NI channels are expected to be deenergized for a period of about 1 minute.

Use of jumpers as directed in 2ICP 11.467 does not constitute a temporary modification, since the reactor protection racks are not in service during the time the procedure is being performed. (SER 92-082)

41. 2ICP 11.469 (Unit 2), (Minor), I&C Surveillance Test Safeguards System Logic - Post As-Built, Revision 0, dated October 22, 1992. (New Procedure)

Summary of Safety Evaluation: Procedure tests control circuits within the safeguards system relay racks as a verification of operability following as-built wire tracing. Analog inputs for steam line pressures and flows, loop temperatures, pressurizer pressure, and steam generator levels are simulated to cause the operation of the logic relays. Logic testing is performed using the existing logic relay test switches in the same manner as specified in ICP 2.19. Verification testing of annunciator, status, and computer inputs requires testing in addition to that specified within ICP 2.19. Parallel and series contacts to initiate alarms and status require close observation of each switch actuation and relay operation to properly document each output. The testing is performed before leaving cold shutdown to ensure operability. ORT 3 testing verifies the output circuitry. Continuity testing is performed on all affected safety injection (SI) circuit components again to assure operability.

The SI block and reset function is tested and verified throughout the test and, in conjunction with ICP 10.1, is used to ensure an actual SI actuation does not occur. Through the testing, the ESF system is not to be considered operable with the exception of the containment ventilation isolation system. (SER 92-083)

42. IT-600 (Common), (Major), Waste Gas System Gaseous Leak Checks (Annual), Revision 9, January 3, 1992. (Permanent)

Summary of Safety Evaluation: The change reduces the waste gas system segregation required to perform the leak testing, thereby reducing abnormal system lineups.

NUREG-0578, recommends leak rate and reduction programs be implemented, and the Lessons Learned section states a position that applicants and licensees will implement leak rate and reduction programs. Point Beach response to the NUREG was a program categorizing systems by likelihood of receiving radioactive fluids from the containment under post-accident conditions. The waste gas system was determined to be a Category 2 system and was to be tested as an ALARA-type test only. Category 2 systems were not to be tested for high source term criteria which made the ALARA-type leak test to atmosphere all that was required.

Since the system is not considered under high source term criteria and a leak test to atmosphere is sufficient, any gaseous leakage undetectable by visual indication is insignificant. (SER 92-007)

43. IWP 88-018*D03, Unit 1, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 1F00962, Revision 1, dated March 27, 1992. (Permanent)

IWP 88-019*D03, Unit 2, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 2F00962, Revision 1, dated March 27, 1992. (Permanent)

The revisions address the decision to enter an LCO during MRs 88-018 and 88-019.

Summary of Safety Evaluation: During the sensor capsule changeout and calibration, the transmitter is isolated from the spray piping. The instrument root valves are red tagged closed. Assurance of the isolation ability of the valves is given by testing them for leakage. This is done prior to the modification. The filling and venting of the instrument is done during an LCO for the "A" train containment spray so containment integrity is maintained during filling of the lines. Filling the lines cannot be done while the instrument is isolated from the spray piping. During the LCO, containment isolation is provided by closing isolation valve 1(2)SI-868A.

The 1SI-868B does not have the special test connection which was installed on the Unit 2 valves during U2R17. The Unit 1 valve is tested as a boundary valve during the ORT testing of the pump discharge check valve. Although this test applies pressure to valve 868B in the reverse direction, it is an air test that does test the overall integrity of the valve. The valve is pressure tested while the transmitter isolation valves are being leak tested. An individual is stationed in containment during this pressure test to verify no water leakage through valve 868B. (SER 91-027-08)

44. IWP 88-018*D-03, Unit 1, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 1F00962, Revision 1, dated March 27, 1992. (Permanent)

The revision addresses the decision to perform MR 88-018 during refueling U1R19.

Summary of Safety Evaluation: The range change involves replacing the transmitter sensing capsule to allow the transmitter span to be increased, allowing the indicating range to be changed from 0-1320 gpm to 0-1800 gpm. The 1SI-868A containment isolation valve for the "A" train containment spray was shut during the filling and venting of the instrument sensing lines because the 1SI-868A valve must be shut or capable of being quickly shut to provide containment closure capability during this step. (SER 91-027-10)

45. IWP 88-018*D-04, Unit 1, Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 1F00963, Revision 1, dated March 27, 1992. (Permanent)

IWP 88-019*D-04, Unit 2, Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 2F00963, Revision 1, dated March 27, 1992. (Permanent)

The revisions address the decision to enter an LCO during MRs 88-018 and 88-019.

Summary of Safety Evaluation: During the sensor capsule changeout and calibration, the transmitter is isolated from the spray piping. The instrument root valves are red tagged shut. Assurance of the isolation ability of the valves is given by testing them for leakage. This is done prior to the modification. The filling and venting of the instrument is done during an LCO for the "B" train containment spray so containment integrity is maintained during filling of the lines. Filling the lines cannot be done while the instrument is isolated from the spray piping. During the LCO, containment isolation is provided by closed isolation valve 1(2)SI-868B.

The 1SI-868B does not have the special test connection which was installed on the Unit 2 valves during U2R17. The Unit 1 valve is tested as a boundary valve during the ORT testing of the pump discharge check valve. Although this test applies pressure to valve 868B in the reverse direction, it is an air test that does test the overall integrity of the valve. The valve will also be pressure tested while the transmitter isolation valves are being leak tested. An individual is stationed in containment during this pressure test to verify no water leakage through valve 868B. (SER 91-027-09)

46. IWP 88-018*D-04, Unit 1, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 1F00963, Revision 1, dated March 27, 1992. (Permanent)

The revision addresses the decision to perform the modification during refueling U1R19.

Summary of Safety Evaluation: The range change replaces the transmitter sensing capsule to allow the transmitter span to be increased. This allows the indicating range to be changed from 0-1320 gpm to 0-1800 gpm. The 1SI-868A containment isolation valve for the "A" train containment spray is shut during the filling and venting of the instrument sensing lines because the 1SI-868B valve must be shut or capable of being quickly shut to provide containment closure capability during this step. (SER 91-027-11)

47. IWP 88-097*A (Unit 1), (Minor), RHR, SI and CS Test Lines Installation, Revision 0, dated March 13, 1992. (New Procedure)

Summary of Safety Evaluation: The IWP installs the system tie-ins for the full flow test lines for the safety injection (SI), containment spray (CS), and residual heat removal (RHR) systems. The pipe routing and support installations are performed under a separate installation work plan (IWP) and safety evaluation. The IWP performs the tie-in of the test line to the discharge cross-tie header of the SI pumps, to the discharge headers of the CS pumps, and to tie-in the return line to the refueling water storage tank (RWST). The IWP performs tie-in of the RHR test line to the discharge headers of the RHR pumps, and tie-in the test line to valve 1RH-742, where the test line flow can be returned to the RWST.

The work is performed with Unit 1 in a cold or refueling condition. Precautions include: no rigging from supports on operable systems; and rigging the pipe being installed in all three directions, if necessary, to assure that there is essentially no possibility of impacting adjacent operable equipment during a seismic event. In addition, the rigging and general state of the construction is reviewed and documented by engineering personnel each shift.

The "A" train tie-in begins as soon as the cavity is flooded and the upper internals are removed. The "A" train is out of service while the core is being off-loaded and the "B" train tie-in does not begin until the core off-load is complete. The "A" train is tested to verify that the ability to remove decay heat (i.e., normal flow path) is unaffected and the hydrostatic test is performed prior to beginning the core reload. In addition, during all periods when fuel is in the vessel and one train of RHR is out of service for this modification, the risk of losing the operable train is minimized by assuring both RHR pumps are operable. (SER 91-074-11)

48. IWP 88-136*C3 (Unit 1), (Minor), Control Room Operator Console Replacement, Revision 1, dated March 9, 1992. (Permanent)

The console replacement was originally scheduled to be performed during normal operations. To address a concern to ensure safe operation during this installation, no work was performed between 0700 and 0800 nor during shift turnovers.

Summary of Safety Evaluation: The two concerns addressed are the ability to adequately monitor the Unit 1 conditions, and the ability to perform EOPs if a situation should arise.

During the Unit 1 installation, the control operator does not have access to the SAS and PPCS terminals that normally reside on the console. However, they are able to use the C03 display, the 2 terminals in the ASIP, the analog trend recorders in the ASIP, and the

Unit 2 PPCS terminal (only if absolutely required). The alternate displays provide adequate monitoring. (SER 89-122-02)

49. ORT 3. (Major). SI Actuation with Loss of AC, Revision 24, dated May 8, 1992 (performed May 14, 1992, Unit 1). (Temporary)

Summary of Safety Evaluation: The position of the service water (SW) pump breakers was changed from the "test" position to the normal "racked in" position. This enables a faster recovery of the SW system. A caution statement was added to effectuate a timely repowering of the residual heat removal (RHR) pump under test should the operating pump cease to function as required by TS 15.3.1.A.3.b (2).

The auxiliary feedwater and service water pumps are unit shared components which are "at power" during the test. The 1P10A RHR pump, P38A auxiliary feedwater pump (AFWP) and P32A&B service water pumps have a sequenced automatic actuation from a Unit 2 safety injection signal and their availability is governed by TS 15.3.3.D (service water system) and TS 15.3.4.A.2 (auxiliary feedwater system). The most limiting condition occurs during UV testing when with P32F service water pump already out of service, the P32A&B pumps are not immediately available due to the "dead bus" condition of 1B03. P38A AFWP is also in the same condition at the same time. Reducing the number of operable service water pumps to three, places Unit 2 in a 24-hour LCO. The test is performed commencing at Step 5.1 of ORT 3 and ending at Step 5.12 when the Unit 1 safeguards buses (4160 & 480 V) are verified powered from their normal supply and by adding both emergency diesel generators in the normal standby mode.

ORT 3, Section 5.10 tests all containment isolation valves that receive a "A" and "B" train trip signal, by verifying that each single train signal independently shuts the valve. With the installation of 1CV-313A and 371A, the dual train actuation of 1CV-313 and 371 was modified to single train actuation via a system modification. This single train actuation is verified in the "Loss of AC Followed by Auto SI" section of the test. (SER 88-060-01)

50. PBNP 8.4.2. (NNSR). Primary Water Chemistry Monitoring Program, Revision 10, dated April 24, 1992. (Permanent)

Summary of Safety Evaluation: The revision increases the "expected range" of component cooling chlorides and fluorides from 150 ppb to 500 ppb.

As with any ferrous system, corrosion occurs, even though the rates may be very small. As a result, an increase in ionic impurity concentration may increase the rate or corrosion. If the rates of corrosion are higher than expected resulting in small amounts of leakage, the corrosion is detected via makeup water requirements and chemical addition requirements. Leakage into the system would be detected by routine radiochemical analysis of the system. Furthermore, inspection programs would also detect small failures.

Larger failures are analyzed in the FSAR. The FSAR analyses for various faults in the component cooling water system are not affected. Leakage detection methods are not affected, nor the required actions change in the event of system leakage.

This revision also changes the lithium control program to conform to the recommendations of EPRI and Westinghouse. (SER 92-034)

51. REI 40.0. (Minor). Pre-Startup Calibration of The Nuclear Instrument Intermediate and Power Range Detectors. Revision 0, dated December 21, 1992. (New Procedure)

Summary of Safety Evaluation: This procedure improves documentation of startup information. The most serious consequence is the possibility that reactor overtemperature protection would activate later than in the evaluated scenarios of the FSAR. However, this risk remains the same whether the recommended Westinghouse calculation is used or estimates based on personnel experience are used. (SER 92-110)

52. RESP 1.1, (Major), Rod Control System: Rod Drop Testing, Revision 3, dated September 24, 1992. (Permanent)

Summary of Safety Evaluation: FSAR Section 14.2.1 describes pre-refueling, individual rod drops to verify the rods are free from the drive mechanisms. Testing the rods in groups versus individually has no effect on the test results. All rods were dropped from a withdrawn position (>170 steps on Bank D) when the reactor trip breakers were open. The RPI system indicated all rods dropped free of the drive mechanism. The intent of this statement is to ensure all rods are free when the head is lifted to prevent inadvertent rod withdrawal. The method used is sufficient to demonstrate this. Additionally, the rods are observed visually when the head is lifted to ensure none are following. Previous tests re-energized the lift coils after the rod was dropped until such time as the reactor trip breakers were opened. This method may actually provide more assurance that the RCCAs are free because the gripper coils were not re-energized. Technical Specifications required drop testing surveillance be satisfied during BOL testing. (SER 91-082-01)

53. RESP 1.3, (Major), Multi-Map Calibration of NIS Detectors, Revision 3, dated May 8, 1990 (performed December 13, 1991). (Temporary)

Acceptance criteria was changed to allow the use of the existing axial offset constants if they are more conservative with respect to their effect on OTΔT.

Summary of Safety Evaluation: The revision allows existing axial offset constants, which are conservative with respect to the constants determined by the recent multi-map calibration, to remain in effect. These constants reflect the gain of the summing amplifiers which generate the Δ flux input to the OTΔT setpoint generating instruments. Leaving the more conservative constants in place results in a more pronounced Δ flux effect on OTΔT and causes a more conservative OTΔT to be generated. It is not desired to use the axial offset constants determined for N42, N43 or N44 at this time because the change would be in the non-conservative direction and a review of the acceptance criteria and methodology of determining setpoints is desired before these 3 constants would be entered on these 3 channels. Changes in sequence for inputting PPCS constants is irrelevant with regard to safety concerns. The change is conservative with respect to Δ flux indication on the MCB. (SER 91-117)

54. RMP 29C, Unit 1, (Minor), 1A05 Bus and Breaker Maintenance, Revision 0, dated April 17, 1992. (New Procedure)

RMP 29C provides procedural control guiding electrical isolation of safeguards bus 1A05 and restoration after maintenance is completed.

Summary of Safety Evaluation: The RMP aligns normal electrical power using substitute breakers (1A52-61 and 2A52-72) if normal breaker maintenance is not completed within the 24-hour limit; use of the 1B03 - 1B04 tie breaker to restore normal breakers after 1A05 was returned to service is limited to 8 hours as described in the FSAR.

Transformers 1&2X03 and 1&2X04, shall be in service during procedure use to prevent a potential single failure loss of offsite power accident. Emergency diesels generators G01 and G02 shall be operable for the duration of the procedure. Testing of Unit 2

safeguards or reactor protection logic is suspended due to potential effects on service water and auxiliary feedwater system operability.

The 4160 volt breakers may be left in the racked-out position in their respective cubicles at 1A05 since any relaying associated with the cubicles are out of service. All 4160 volt breakers are labeled to assure they are restored to their original cubicles.

Use of temporary jumpers for 480 volt breakers is in accordance with the safety evaluation addressing bypass jumpers used in RMP 23A, Unit 1, "480 V Breaker Maintenance." (SER 91-087-02)

55. RMP 29D, Unit 1, (Minor), 1A06 Bus and Breaker Maintenance, Revision 0, dated April 17, 1992. (New Procedure)

RMP 29D provides procedural control guiding electrical isolation of safeguards bus 1A06 and restoration after maintenance is completed.

Summary of Safety Evaluation: The contingency appendix also provides for a G01 alternative, so that if G01 is operable under the loss of offsite power and diesel loading is satisfactory, P12B can be repowered by tying 1B04 to 1B03 to provide additional SFP cooling capability.

Boric acid transfer pump 1P4B is powered via MCC B43. With 1A06 out of service and an under voltage condition, this MCC strips from 1B04. Therefore, the RMP assures that at least two boric acid transfer pumps are in service with at least one pump lined up to Unit 2.

Testing of Unit 2 safeguards or reactor protection logic during this RMP is suspended due to potential effects on service water system operability. Both emergency diesels generators G01 and G02 shall be operable.

The 4160 volt beakers may be left in the racked-out position in their respective cubicles at 1A06 since any relaying associated with the cubicles is out of service. All 4160 volt breakers described in the RMP are labeled to assure they are restored to their original cubicles.

Use of temporary jumpers for 480 volt breakers is in accordance with the safety evaluation addressing bypass jumpers used in RMP 23A, Unit 1, "480 V Breaker Maintenance." (SER 91-087-03)

56. RMP-48A, (Minor), 1X03 Maintenance, Revision 0, dated April 23, 1992. (New Procedure)

This procedure controls the removal from and return to service of 1X03, the high voltage station auxiliary transformer, and associated breakers and equipment. During the time 1X03 is removed from service, offsite power is provided to both units through 2X03. When 1X03 is removed from and returned to service, 1X03 and 2X03 secondaries are paralleled for a short period of time. The procedure states maintenance shall be performed with Unit 1 in a cold or refueling shutdown condition, with all major motor loads on 1A01 and 1A02 secured and electrically isolated.

Summary of Safety Evaluation: The 13.8 kV system is the supply of offsite power to the site. Technical Specifications allow an X03 transformer associated with a shutdown unit to be out of service indefinitely. If the unit is to be started up without its X03 in service, then the gas turbine shall be operating. Although not specifically required, it is prudent to maintain the gas turbine operable when a single X03 transformer is in service. An

exception is made to maintaining the gas turbine operable as a standby power source during refueling shutdown with the core completely unloaded since refueling cavity cooling is not required.

With 2X03 supplying offsite power to both units, upon a Unit 2 trip and design basis accident requiring ESF actuation, there can be a normal fast bus transfer (2A01 to 2A03, 2A02 to 2A04) of that unit's non-safety-related loads to the station auxiliary transformer (2X04) as well as full loading of the safeguards buses. The single 2X03 transformer, which supplies both 1X04 and 2X04 transformers, is sized to accept this added loading based on a similar loading analysis determined for a single X04 transformer (SER 87-016). The analysis is directly applicable to an X03 transformer due to its equivalent power rating. The analysis assumes normal Unit 2 4160 V loads, fully loaded Unit 1 and Unit 2 480 V buses, and no Unit 1 4160 V loads due to cold shutdown. Therefore, as a precautionary measure to minimize the potential for overloading 2X03, all major 4160 V motor loads on 1A01 and 1A02 are secured and electrically isolated.

While 1X03 is out of service, if major maintenance on 1X03 becomes necessary such that it takes more than 4 hours to restore 1X03, consideration shall be given to removing main generator disconnects in preparation for backfeed per AOP-14A. Also, if the gas turbine becomes inoperable, consideration shall be given to removing the main generator disconnects in preparation for this backfeed. This is based on available power sources (2X03, gas turbine, 1X01/1X02 backfeed) and the desire to maintain reliable power source alternatives. An exception is made to gas turbine operability during refueling shutdown with the core completely unloaded. (SER 92-039)

57. RP-6B, (Major), Steam Generator Crevice Cleaning, Revision 1, dated August 13, 1992. (Permanent)

Caution statements were added before heatup and cooldown steps to specify TS and administrative heatup and cooldown rates. The procedure includes actions to be taken if limits are exceeded and applies the requirements of PBNP 3.4.19 "Infrequently Performed Tests or Evolutions/Special Test Procedures."

The crevice flushing begins at a lower initial temperature, 240-250°F vice 325-335°F to minimize the risk of an excessive cooldown.

Summary of Safety Evaluation: The procedure affects the steam generators and the FSAR describes flushing. TS 15.3.1 and 15.4.2 address steam generators. These descriptions are not adversely impacted by the changes.

The changes clarify methods to better control heatup and cooldown cycles, highlight Technical Specification requirements for decay heat removal and cooldown limits, and add administrative steps to increase the monitoring and control of the evolution. The changes prevent possible adverse impact on reactor coolant system integrity and ensure adequate decay heat removal capability is maintained. (SER 92-063)

58. SMP 1091 (Unit 1), (Minor), 1CC-769 Inspection and Repair, Revision 0, dated January 6, 1992. (New Procedure)

This SMP performs maintenance on containment isolation valve (CIV) 1CC-769 because the valve failed its stroke timing test. This CIV and associated section of piping is isolated by closing the inside containment manual valve 1CC-865B, and the outside containment manual valve 1CC-772. These valves are leak tested and shut. Motor-operated valve 1CC-865B serves as the primary containment isolation barrier during repair of 1CC-769. 1CC-769 is tested to ensure operability prior to returning the valve to service.

Summary of Safety Evaluation: Isolating this section of the component cooling (CC) system removes the excess letdown heat exchanger from service. The rest of the system is functional. Maintenance on this valve while it is isolated from the CC system does not affect any other SSCs important to safety.

Inside containment 1CC-865B, is shut and leak tested to CC system pressure of 120-125 psig, which exceeds containment design pressure. CC is considered a closed system outside containment, and the closed system serves as a redundant isolation barrier for the reactor coolant pump (RCP) seal water penetrations as well as the excess letdown heat exchanger penetrations. To maintain closed system integrity and to isolate 1CC-769 for maintenance, 1CC-772 is shut. This valve is leak tested in the proper direction, however, due to the CC system design, the valve is leak tested to ~22 psig. This is the approximate pressure of the CC return from the RCP seals. To test at a higher pressure requires isolating a major portion of the CC system. The acceptance limit for the leakage test for these valves is 20 ml/hr. (SER 92-001)

59. SMP 1095, (Minor), SW-307A Disc Repair, Revision 0, dated February 17, 1992 (New Procedure)

Summary of Safety Evaluation: SMP 1095 requires operators to reduce the vacuum on the service water return header which normally runs at 20" of Hg to between 1" to 3" Hg to permit bonnet removal at a low vacuum which reduces the rate that air is drawn into the service water header. The concern with the air inleakage is the Unit 1 condenser waterboxes where the air in the service water return header ends up. To preclude this problem, the spare vacuum priming pump is lined up to the Unit 1 vacuum control tank to handle needed air removal.

To reduce air inleakage and limit flooding potential, a wooden disc the same size as the valve disc was fabricated. This disc is placed in the valve body as soon as the bonnet and disc are removed. As a backup to the disc, a valve body cover made of 3/4" plywood is fabricated and bolted on to the valve body if needed. The expected pressure range in the area of the opening can range from ~5" Hg to a slightly positive pressure. Personnel are available to make adjustments to maintain the pressure slightly negative.

Operation of equipment that changes conditions in the service water system such as the starting of extra service water pumps on a Unit 1 or 2 safety injection is addressed. To remove the possibility of the change, four service water pumps will operate along with containment cooler service water valves SW-2907 and SW-2908 open when the conditions for maintenance are established. All six pumps are not required because by the time four service water pumps are running the header discharge pressure is at a maximum of 75 psig and does not go any higher with two more pumps. (SER 92-010)

60. SMP 1100, Unit 1, (Minor), Corrective Maintenance of 1SC-966B with Containment Integrity Established, Revision 0, dated April 2, 1992. (New Procedure)

SMP 1101, Unit 2, (Minor), Corrective Maintenance of 2SC-966B with Containment Integrity Established, Revision 0, dated April 2, 1992. (New Procedure)

The procedure performs corrective maintenance on 1(2)-SC-966A, 966B, or 966C containment isolation valves (CIVs). These are, respectively, pressurizer steam space sample, pressurizer liquid space sample, and hot leg sample valves. These CIVs are located outside containment. The containment isolation features and configurations of these penetrations are similar. Therefore, this safety evaluation bounds the work performed on the valves with the limitation that the steps taken to ensure containment isolation capability throughout the maintenance activity are as described in this safety

evaluation, and in SMPs 1100 and 1101. SMPs 1100 and 1101 control the maintenance done on 1(2)-SC-966B. This corrective maintenance is done during power operation. During this maintenance, the respective sample line is out of service. Containment isolation is provided at all times by closing the Appendix J tested, inside containment CIV.

Summary of Safety Evaluation: The only accident analyzed in the FSAR related to CIV corrective maintenance is the small-break LOCA, should a leak from the pressurizer of hot leg occur. This accident is not applicable because the sample line is a 3/8" line. The FSAR discussion of the small-break LOCA event states that breaks 3/8" and smaller are considered within the makeup capability of the charging system and allows enough time for operator action. The probability of a LOCA is not increased during this maintenance because a fully qualified and tested CIV is used to isolate the valve worked on from the primary system.

The sample lines in question do not act as a support system for other equipment important to safety. This activity does not affect any other equipment important to safety. (SER 92-028)

61. SMP 1104, Unit 1, (Minor), Replace Flange and Reducer Downstream of Valve SW-144, Revision 0, dated April 7, 1992. (New Procedure)

Summary of Safety Evaluation: SMP 1104 coordinates replacement of the reducer and flange. The work requires the Unit 1 fan coolers to be out of service. Thus, the replacement is performed during refueling shutdown or cold shutdown. The reducer location is not isolable from the 20" service water return header. To provide isolation, a freeze seal is utilized which uses liquid nitrogen as a cooling medium. It is considered a safe isolation method provided preliminary pipe examinations are performed and several precautions are taken which include: 1) an NDE inspection of the area to be freeze sealed to ensure no detrimental defects exist that could initiate brittle fracture, 2) supplementary pipe supporting in the area of the freeze seal to minimize dead weight stresses, 3) contingency plans to be taken in the event of a loss of the freeze seal, 4) provisions for inhabitability of the freeze seal work area if the concentration of nitrogen gas gets too high, and 5) use of personnel with expertise in freeze sealing.

The NDE aspects performed via UT and MT inspections of the pipe. Additional support is provided by jack stands in the vicinity of the freeze seal. Contingency plans for the loss of the freeze seal include fully opening of the overboard valve in the event of a Unit 2 SI while the overboard pressure is positive to provide conditions necessary to form and maintain the freeze seal; securing of service water pumps if the freeze seal is lost or the pipe breaks at the freeze seal location; having an inflatable stopper available for inserting into a hole unisolated from the 20" return header; and having a 14" slip-on and blind flange at the job site for subsequent installation for a substantial isolation means. In addition to having a freeze seal for isolation, an inflatable stopper is inserted into the pipe, during as much of the evolution as possible, upstream of the freeze seal when the system is opened. The stopper handles as much as 15 to 20 psig line pressure. Regarding displacement of oxygen in the PAB due to the nitrogen exhausted from the freeze chamber, a fan is set up for dilution, air monitors are used to monitor air quality, and if possible, a hose is set up to route the nitrogen gas directly to the PAB ventilation system. SCBA equipment is available at the location.

Protection against PAB flooding is of utmost importance since there are numerous pieces of important to safety equipment (boric acid transfer pumps, SI pumps, spray pumps, electrical panels, etc.), in the immediate area. NRC IN 91-041 "Potential Problems With The Use Of Freeze Seals" was reviewed and weaknesses were addressed. (SER 90-056-02)

62. SMP 1114, (Major), Functional Test of Miscellaneous Breakers and Circuits Following As-Built Wire Tracing in C02, Revision 0, dated October 23, 1992. (New Procedure)

SMP 1114-1, (Minor), Functional Test of Miscellaneous Breaker and Circuits Following As-Built Wire Tracing in C02, Revision 0, dated October 23, 1992. (New Procedure)

Summary of Safety Evaluation: A technical review included Technical Specifications, FSAR, associated drawings, and human factoring. Breaker circuits that run through panel C02 are tested by voltage checks, placing breakers in local control, then making circuit continuity checks using the main control board switch, and by using continuity checks at the breaker cubicles with the breaker still racked-out. When testing 2B52-38B, 2B03 Feed To MCC 2B-32, the breaker is first placed in local control so power is interrupted to safeguards MCC 2B-32. Testing is then performed by panel C02 switch manipulation. This test is performed when the reactor is in cold shutdown. Steps were added listing conditions necessary to verify the equipment has been returned to service. (SER 92-096)

63. ST-2, (Major), Core Cooling, Revision 2, dated October 11, 1991 (performed February 4, 1992). (Temporary)

Summary of Safety Evaluation: The wide range RVLIS was modified by MR 88-018*K so the reactor vessel wide range level block was changed. The values currently listed are more conservative adverse containment setpoints. The normal containment setpoints are added and the adverse containment values enclosed in brackets as for other setpoints. Narrow range values are not affected and adverse containment values are not necessary since the calculations show that the normal and adverse values are the same. (SER 89-041-02)

64. ST-2, (Major) Core Cooling, Revision 3, dated February 24, 1992. (Permanent)

Summary of Safety Evaluation: The wide RVL was modified by MR 88-018*K so the reactor vessel wide range level block was changed. The values currently listed are more conservative adverse containment setpoints. The normal containment setpoints were added and the adverse containment values enclosed in brackets as for other setpoints. Narrow range values are not affected and adverse containment values are not necessary since the calculations show that the normal and adverse values are the same. (SER 89-041-03)

65. STPT 2.4, (Major), Containment Fan Safeguards Sequence Time Delay Relays Setpoint Change, Revision 0, dated November 12, 1992. (New Procedure)

Summary of Safety Evaluation: The time delay relay (TDR) setpoints which control the safeguards sequencing of the containment fan cooler motors following a safety injection initiation were changed. The setpoint change for TDR-16 and TDR-26 changes from 30 seconds to 40 seconds. The setpoint change for TDR-17 and TDR-27 changes from 35 seconds to 45 seconds. There is no interim configuration or conditions.

The setpoint change ensures the consequences of an accident remain within the bounds of the analysis by assuring the initial assumption of no containment fan cooler prior to 35 seconds is satisfied. The change to the TDRs has no affect on the relays physical operation or the subsequent starting of the containment fan cooler motors. Since the hardware was not replaced or physically modified, the probability of an occurrence of a malfunction is not altered. (SER 92-050)

Summary of Safety Evaluation: The setpoint changes the safeguards sequence time delay relays TDR-17 and TDR-27, which control the safeguards sequencing of the containment fan cooler motors 1&2W1B1 and 1&2W1D1 respectively, following a safety injection initiation from 35 seconds to 50 seconds. There would be no interim configurations or conditions.

A complete evaluation of the basis for the settings of the TDRs for the containment fan coolers and the other electrical loads, which are started by the safeguards sequencer, was performed. (SER 92-089)

66. TS-31 (Unit 2), (Major) High and Low Head Safety Injection Check Valve Leakage Test (Cold Shutdown), Revision 11, dated November 4, 1991. (Permanent)

Free convection is the only means of removing heat from the core when core deluge is used. Calculation N-91-112 verifies that free convection is an adequate means of decay heat removal provided fresh assemblies are loaded and the core has been shut down for at least 20 days. Free convection currents are expected to create down flow in the fresh assemblies and up flow in the burned assemblies. The procedure was changed to ensure adequate decay heat removal is available by adding precautions and limitations, and additional instructions to monitor temperatures.

Summary of Safety Evaluation: The test satisfies the requirements of TS 15.3.1.A.3.b.4. This specification states that one of the two residual heat removal loops may be temporarily out of service for surveillance requirements, provided it can be returned to service if needed. TS-31 operates one pump and one heat exchanger with injection into the reactor coolant system (RCS) using the core deluge lines instead of normal RHR flow path into the cold leg. Using the core deluge lines does not provide forced circulation through the core. Decay heat removal must take place using free convection during the test. Inadvertent boron dilution is prevented by shutting valve CV-360 which provides redundant isolation of reactor makeup water from all injection paths to the RCS. (SER 91-118)

DESIGN CHANGES

The following modifications were installed prior to 1992 but were inadvertently omitted from previous reports.

1. MR 85-048*T (Common), Buildings and Structures. MR 85-048*T installs a north service building (NSB) to house a new water treatment plant, training facilities, storeroom, secondary chemistry lab, and expanded staff.

   Summary of Safety Evaluation: The existing 13.8 kV electrical system provides offsite power from high voltage station auxiliary transformers 1&2X03 to low voltage station auxiliary transformers 1&2X04. The H01 switchgear allows a cross-tie between units and tie-in of the gas turbine. The physical separation and isolation of the new equipment greatly increases the reliability of the 13.8 kV system. The new switchgear provides separate feeder breakers to provide power from the 13.8 kV system for the new NSB. (SER 87-022)

2. MR 85-056 (Unit 1), MR 85-057 (Unit 2), Main Control Boards. The modifications replace the subcooling meter analog display with a digital display for post-accident monitoring as required by NUREG-0737.

   Summary of Safety Evaluation: The displays utilize red LED segments of the same size and type used in other control room displays and are easily read by the operators. The

digital displays require 120 Vac, which the analog displays did not require. One will be powered through a 10 amp breaker from the white instrument bus and the other through a 10 amp breaker from the yellow instrument bus. The digital displays require 6 amps power each, which is a negligible increase in load on each bus. (SER 90-041-02)

3. MR 85-214*A, (Unit 2), Anticipated Transients Without Scram (ATWS) Mitigating System Actuation Circuit (AMSAC). AMSAC is designed to trip the main turbine and start the motor-driven and steam-driven auxiliary feedwater pumps when a loss of main feedwater flow is detected.

   Summary of Safety Evaluation: Loss of main feedwater flow is detected by sensing both main feedwater pump motor circuit breakers are open or both main feedwater control valves are closed. AMSAC performs its function when the unit is above 40% power. The output signals are internally delayed by 30 seconds to allow the reactor trip circuits to actuate prior to AMSAC.

   AMSAC is classified as non-Class 1E, but interfaces with the auxiliary feedwater pump start circuits which meet Class 1E requirements. The interface and isolation are provided by output relays. These relays are mounted in the AMSAC enclosure and were purchased as Class 1E seismically qualified relays. In addition, they were qualified to Appendix A per NRC SER letter dated September 17, 1986. The AMSAC enclosure and the conduit carrying the cables to the auxiliary feedwater circuits are seismically mounted.

   The switches installed on the main feedwater regulating valves for AMSAC are the same type as the previous valves and are environmentally and seismically qualified. Connectors mounted on the new switches were ordered with equivalent qualification because of high energy line break considerations. Environmental and seismic qualifications for these switches are not design requirements for AMSAC.

   AMSAC is not redundant and has only one source of power. Power to AMSAC is through breaker panel 1&2Y06. The source is normally offsite power with a diesel generator backup. Upon loss of offsite power, the diesel has power restored within 10 seconds. (SER 89-055)

4. MR 87-034*E, *F, and *J (Common), 480 V Electrical System. This modification replaces the existing 480 V BD breaker electro-mechanical overload protection with solid-state overload protection. The new protection consists of three sensors, one Amptector solid-state trip unit, and one direct trip actuator for each breaker.

   Summary of Safety Evaluation: All units on safety-related buses (B03/B04) are qualified for seismic installation per IEEE-344. Installation was in accordance with the manufacturer instructions and addenda for retrofitted BD breakers. This assures problems experienced with the direct trip actuator as described by NRC IN 88-054 are corrected. For safety-related breakers, full current testing is performed to ensure all settings are correct and the new overload protection functions as expected. (SER 88-137)

5. MR 87-121*C and *K (Common), 480 V Electrical System. The modification provides a dedicated source of 480 V power to selected plant equipment independent of the 4 kV switchgear room. The modification meets the requirements of 10 CFR 50 Appendix R Section 3L for a dedicated shutdown system, when physical separation criteria cannot be met.

MR 87-121*C installs the alternate shutdown substation and MR 87-121*K installs the cable pulls to the alternate shutdown transformer.

Summary of Safety Evaluation: The design packages ensure train separation and/or separation of normal/alternate sources is provided and maintained, and safe shutdown operability is provided for specific plant equipment in accordance with 10 CFR 50 Appendix R. The interface between the alternate (dedicated shutdown) supplies and existing Class 1E equipment is via Class 1E switching devices. Equipment and raceways associated with the alternate supplies are seismically installed in seismic plant areas. The switching devices utilize mechanical interlocks such that the normal and alternate sources cannot be tied together. It is possible to deenergize both normal and alternate 480 V supplies to a given piece of equipment by switching at the local switch panels. Although not required based upon criteria of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System," control room annunciation is provided to alarm abnormal switching alignments.

Implementation of this modification does not affect the present operation of existing plant equipment/systems when switching devices are in their normal alignment. A worst case failure of equipment associated with the dedicated shutdown system requires fire watches in one or more plant fire areas. Tie-ins to existing plant systems and startup testing for this modification are controlled by a special maintenance procedure. (SER 87-054)

6. MR 87-217*C (Common), Main Control Boards. MR 87-217*C provides control room indication for SA-9. SA-9 control room indicator is wired in parallel to the local SA-9 indicator in the evaporator control panel.

Summary of Safety Evaluation: The addition of indicating lights (and consequential cutting of holes) in main control boards 1C03 and 2C03 does not adversely affect the seismic qualification of the control boards. Protection for the addition of remote wiring and control room wiring and indication is provided by the fail safe control circuits of the SA-9 valve. SA-9 fails safely in the closed position, providing isolation of the radwaste system. This modification does not affect operation of SA-9. (SER 90-110)

7. MRs 87-227*A, *B, and *C (Common), Electric Generator/EDG. MRs 87-227*A, *B, and *C reverse the EDG voltage regulator and governor control switches and main generator, and the minimum and maximum excitation lamps, which are located directly above the base adjust controls. The modification resolves HED 57, control room design review.

Summary of Safety Evaluation: Control switch changes involve the exchange of two wires at each switch, and the installation of some nameplates. Indication lamp changes involve exchanging one wire from each lamp, and exchanging the lamp labels.

Null meters for each unit are rewired so null meter motion follows the motion of the modified base adjust switch. Since discrepancies were discovered between the installed null meter wiring and the plant drawings, rewiring of the meters to match the drawings returns the circuits to original design.

Independent verification is performed to verify leads as they are lifted and reterminated. Continuity checks are used to verify proper switch operation and the integrity of the reterminated leads.

There are no seismic or Appendix R requirements and no system functional changes. (SER 90-048)

8. MR 88-012*C and *E (Common), Circulating Water System. The design packages install connections for the chlorination systems.

Summary of Safety Evaluation: Work is performed in accordance with B31.1-1967 and approved plant welding procedure WP-4.

Each penetration on the outboard side of the condensers is closed off by an isolation valve and a threaded cap. This precludes any flooding concern or significant waterbox air in-leakage that could cause the condenser waterbox level to drop. (SER 88-044)

9. MR 88-093*A, *D (Unit 1), 88-094*A, *D (Unit 2), Chemical Injection System. The design packages install hydrazine injection conduit and cable and mechanical portions of the modification.

Summary of Safety Evaluation: The design packages do not change the chemical addition point into the condensate system, nor any secondary system water chemistry action levels. Therefore, corrosion rates of steam generator tubes are not negatively impacted.

A malfunction of the hydrazine controller would be detected by the residual hydrazine, oxygen or pH analyzers. The abnormal levels cause secondary sample panel alarms which bring in the control room secondary sample panel alarm. With the problem in hydrazine feed rate identified, the pumps could be placed in manual operation if necessary to repair or adjust the controller.

The Unit 1 system is supplied with power from 480 V power panel PP-3, while the Unit 2 system is powered from 480 V power panel PP-8. The load analysis showed the additional loads being placed on the power panels is acceptable. These power panels are not supplied by the emergency diesel generators. (SER 89-103-02)

10. MR 88-098*D, IWP 88-098*A1, IWP 88-098*B1 (Unit 2), Safety Injection System. MR 88-098*D installs piping and supports for the residual heat removal (RHR), safety injection (SI) and containment spray (CS) full flow test lines. Interim configurations include the temporary supporting of these new piping systems during construction with both units at power.

Summary of Safety Evaluation: There is no impact on existing systems. All system tie-ins and modifications to existing system supports are performed during U2R17.

The modification is performed in accordance with B31.1-1967. The installation uses standard rigging practices for erection of piping. To assure the interim configurations are adequate from a Seismic 2/1 concern and that there is no effect on existing system or supports, no rigging is allowed on existing piping or piping supports. All rigging is done from the new supports or from lifting lugs which are designed and installed under MR 89-056*A.

The rigging uses a minimum safety factor of 5 beyond rated capacity and prevents swinging of rigged piping. This prevents the possibility of damaging adjacent systems. In addition to these controls, an engineering inspection is performed at the end of each shift to verify the adequacy of the interior support configuration. The review considers possible impacts on adjacent piping and supports due to falling or swinging piping or supports. There is no affect on the adjacent systems as a result of this installation. (SER 91-074)

11. MR 88-099*A (Unit 1), Auxiliary Feedwater System. MR 88-099*A adds flow measurement instrumentation to the recirculation line and increases the capacity of

1P29 auxiliary feedwater pump. The modification was initiated in response to NRC Bulletin 88-004 as refined by NRC Generic Letter 89-004.

Summary of Safety Evaluation: The capacity of the mini-recirculation lines is increased from 30 gpm to a minimum of 100 gpm based on the manufacturer's recommendations. To meet the requirements of ASME Section XI testing, flow indication is provided on the recirculation lines.

Calculation N-91-032 estimated flow to the steam generators may be reduced to 162 gpm per steam generator if this failure occurs with a steam generator pressure of 1100 psig. The 100 gpm per steam generator required in the accident analyses is still available. In addition, manual valve AF-15 can be used to isolate the recirculation line or air to the control valve can be isolated. Calculation N-91-007 verified that there is more than 5 minutes for manual action to isolate the recirculation line. AF-4002 has position indication in the control room so this failure can be identified. AF-4002 was added to the ASME Section XI test program and is verified to open and shut per inservice test IT-290.

The modification for 1P29 was completed during U1R19. The isolation for this work results in the recirculation lines for pumps P38A, P38B, and 2P29 being out of service in addition to the 1P29 recirculation line. P38A&B and 2P29 pumps are not out of service since discharge paths to the steam generators are lined up to compensate for the isolation of the recirculation lines. Administrative controls were established to minimize the potential for damage to the P38A&B and 2P29 pumps during the time that the tie-in to the common recirculation discharge line was installed for 1P29. (SER 91-025)

12. MR 88-172 (Unit 2), Instrument Air. The modification upgrades the instrument air supply to the crossover steam dump valves in accordance with NRC Generic Letter 88-14 by installation of a check valve on the main supply line to the crossover steam dump valves and connecting an accumulator downstream of the check valve. Accumulator volume is sufficient to open the four dump valves.

Summary of Safety Evaluation: The modification ensures instrument air is available to operate the crossover steam dump valves. Although originally designed to operate with a loss of instrument air, some of the valves failed to open when tested on a loss of instrument air.

The check valve has a 5 psi spring to maintain pressure on the air lines to the dump valves and in the accumulator. The accumulator is sized to maintain sufficient pressure while operating the valves per calculation N-90-055. With the accumulator and check valve, the crossover steam dump valves remain functional even with a loss of instrument air system pressure. This increases the functional reliability of the crossover steam dump system without creating additional possible malfunctions or accidents. The crossover steam dump system is designed to prevent overspeed of the low pressure turbines. The modification increases the reliability of the system. (SER 90-098)

13. MR 88-188*A (Common), Miscellaneous Motor-Operated Valves (MOVs). The MR installs replacement 4-rotor limit switches, and provides overload trip indication on motor-operated valves.

Summary of Safety Evaluation: The operating time and characteristics for valves does not change with the introduction of the new 4-rotor limit switch. This change enhances MOV position indication and torque switch bypass operation only. The function and operation of interlocks remains the same.

Overload indication in the control room is discussed in FSAR Section 7.5.2. The modification makes the few non-overload indicating MOVs consistent with the FSAR. The ground wire of the open and shut light was moved to the positive terminal of the overload relay so both indication lamps are deenergized on MOV overload. The overload relay contacts are adequate for the additional current of the indication lamps. Wiring changes were made in the motor control center.

No new penetrations were made; there were no additional loading of conduit or cable trays. There are no Appendix R considerations associated with these changes. (SER 90-039)

14. MR 88-188*E (Unit 2), Miscellaneous Motor-Operated Valves (MOVs). MR 88-188*E installs T-drains on the motor-operated valves installed in Unit 2 containment and safety-related valve operators installed outside of containment.

Summary of Safety Evaluation: Adding T-drains to safety-related or non-safety-related valve operators provides additional protection from moisture intrusion. The manufacturer states the addition of the T-drains does not adversely affect the EQ rating of an MOV. This modification has no affect on valve motor-operated structure, qualification or function. (SER 90-039-01)

15. MR 89-023-02, (Unit 1), Reactor Coolant System. MR 89-023-02 modifies the reactor vessel (RV)flange leakoff drain lines to the reactor coolant drain tank (RCDT).

Summary of Safety Evaluation: Manual valve RC-522, which is the final valve to the RCDT from the RV flange leakoff line, was moved upstream approximately 15' to alleviate ALARA concerns. In addition, the 3/8" tubing line between the RV flange leakoff line and manual valves RC-522A, 523, and 525 which allows the RV flange leakoff line to be drained through LT-447's available leg drain valve (RC-525A) is eliminated, and a 3/8" drain valve is added to the RV flange leakoff line downstream of temperature element TE-418. The addition of this drain valve allows the drainage of the RV flange leakoff line to be visually verified without affecting LT-447 or LT-447A, and allows procedure RP-1B to be simplified, thus reducing the chance of valve mispositioning. These installations involve 3/8" Swagelok fittings and are QA and seismic. A maintenance work request was used to control work and post-installation leak check performed as practical to 2500 psig. Since the modified portion of the RV flange leakoff line is downstream of temperature element TE-418, this change does not significantly impact indication of RV flange leakoff. (SER 89-054-01)

16. MR 89-034*C (Common), Main Steam System. MR 89-034*C installs a 1-1/2" header between the component cooling heat exchanger room and the Unit 2 turbine hall. This line provides a flow path to route the steam drains in the area to the Unit 2 LP trap header.

Summary of Safety Evaluation: The modification is installed per the original design and subsequent enhancements of the affected systems. The steam inlet is designed to 1085 psig with carbon steel materials. The steam trap outlet is 1085 psig but uses stainless steel. Use of stainless steel at the discharge alleviates most of the concern with erosion/corrosion.

The steam trap assemblies installed upstream of MOV-2019 and 2020 are installed so they do not affect the seismic rating of the system. The first valve on the 3/4" line from the 3" auxiliary feedwater pump (AFP) steam line is designed to be the seismic, QA, and safety-related boundary. This was done so the installation has a negligible effect on the seismic performance of the existing system. This is the same concept applied to similar steam trap applications throughout the plant. Such is the case with the steam trap

assemblies which trap the main steam lines upstream of the MSIVs and the non-return valves. Based on these reasons, the likelihood of a failure of the line is not increased and therefore, the possibility of a steam line break accident is not increased. Also, the possibility of a failure of the AFP or the steam line as a primary heat sink or a containment boundary is not increased.

If a failure occurs, its consequences is limited by the consequences of a failure of the existing system. A 3/4" steam line break is well within the bounds of the current FSAR analysis for a secondary line break. Also, this MR adds new steam and condensate lines in the vicinity of the boric acid storage tanks (BASTs) and MOV-2019/2020 valves. The EQ analysis of this area already assumes a steam environment due to a line break. Addition of the 3/4" steam trap lines does not affect this. From a pipe whip or high energy line break standpoint, the 3/4" line is assumed in our analysis to not affect anything of a larger nominal size. A review of the layout has also shown that no specific safety-related components are affected by a break in these lines. Therefore, EQ and HELB are not a concern. (SER 90-087-02)

17. MR 89-040. Security. This modification contains safeguards information.

18. MR 89-046. Buildings and Structures. MR 89-046 installs a backdraft damper into each of the turbine buildings to provide a passive mechanism to relieve flooding that could result from a catastrophic circulating water seal failure.

    Summary of Safety Evaluation: Both dampers are installed during normal plant operation and requires penetrating a security barrier. Security is contacted prior to penetrating the barrier and only one damper is worked at a time to avoid penetrating two barriers simultaneously. The dampers are installed so they cannot be removed from the outside to prevent uncontrolled access.

    The slight modification of the structural steel has no affect on the structural integrity of the turbine building structure. (SER 90-067-01)

19. MR 89-077*A (Unit 1) and 89-078*A (Unit 2). Crossover Steam Dump System. MRs 89-077*A/078*A install thermometers on the crossover steam dump stacks to determine if the valves have actuated.

    Summary of Safety Evaluation: All components are adequate for Seismic Class 2 qualification so the crossover steam dump valves and surrounding equipment are not impacted during a seismic event. The existing crossover steam dump system and the gauges are not seismically qualified and are non-QA Scope. There are no Appendix R concerns. This installation does not affect the operation of the steam dump system. (SER 90-033)

20. MR 89-188*D (Unit 1). 125 V DC System. MR 89-188*D replaces existing Type HFA breakers D13, 2, 4, 14 and 16 with Type RHFA breakers having thermal-magnetic trips.

    Summary of Safety Evaluation: D13 breakers 2, 4, 14 and 16 supply normal control power to 2A01, 2A02, 2B01, and 2B02. The replacement breakers have equivalent ratings, and the thermal-magnetic trip elements have an instantaneous trip feature on high fault currents. This provides isolation between non-safety-related buses 2A01, 2A02, 2B01, 2B02 and the safety-related DC distribution system. A single fault on a non-safety-related system does not cause a loss of function of the safety-related DC supply.

The use of thermal-magnetic trip breakers in D13 may not provide the best coordination with downstream fuses used in breakers A01, A02, B01 and B02. A fault could occur in one breaker cubicle which should blow the supply fuse, but instead may cause the bus supply DC circuit breaker to trip instead. This prevents automatic trip, including overcurrent trip, of the breakers on these buses, due to a loss of control power. Manual trip of these breakers is still available. Automatic trips associated with these breakers are: undervoltage; bus lockout; feedwater isolation trip to main feed pump; condensate pump and heater drain tank pump motor breakers; reactor coolant pump underfrequency trip; and main feed pump trip on low oil or low suction pressure.

A single fault associated with the DC distribution does not prevent automatic feed isolation since the main feed regulating valves shut in such a scenario.

The single fault does not prevent reactor trip on 2A01/2A02 underfrequency to reactor coolant pump trip above P8. Below P3 reactor trips is dependent upon low flow trip. This is no different than the current case of loss of DC control power to a single non-vital 4 kV bus. Steam-driven auxiliary feed pump initiation on opening of both main feed pump breakers on UV is lost. The motor-driven auxiliary feed pumps provide a backup. (SER 89-134-05)

21. MR 89-191 (Unit 1). Residual Heat Removal System. The modification reconfigures Unit 1 residual heat removal (AC-601R-2) relief valve 861C discharge. The relief valve discharge was installed to connect to the El. 21' floor drain system. These drains then discharge into sump A.

Summary of Safety Evaluation: The change allows seismically qualifying the line for decay heat removal use until the piping configuration can be completely corrected.

The installation is performed in accordance with B31.1 requirements. The revised configuration, including the 10" AC-604R-2 residual heat removal piping, was reviewed for design loading including pressure, deadweight, seismic and relief valve discharge. (SER 89-138-01)

22. MR 90-074 (Unit 1). Condensate System. MR 90-074 installs impingement grating which deflects turbine steam exhaust, to protect condenser tubes from steam erosion. The general location of the grating is near the top of the tube bundles and along the outer tube bundle perimeter. All materials are 304 stainless steel and are securely fastened.

Summary of Safety Evaluation: Installation is performed during a refueling outage. The grating is located where tube erosion was identified by both eddy current and visual inspections. Heat removal capability of the condenser is unaffected because the surface area of the tubes is unchanged and the flow area is not appreciably reduced by the grating. This design concept has effectively controlled erosion at other facilities.

This modification does not affect the Seismic Class 3 qualification of the condenser.

The safety concerns from an FSAR accident analysis viewpoint are not applicable. The condenser is not relied upon to perform heat removal under analyzed accident conditions; the steam generator safety valves are relied upon to perform this function. (SER 90-088-02)

23. MR 90-226*B (Unit 2). Chemical and Volume Control System. MR 90-226*B replaces volume control tank (VCT) level transmitters, LT-112 and LT-141.

Summary of Safety Evaluation: The control and alarm functions of the VCT level transmitters are described in the FSAR; however, there are no specific details which actually describe the transmitters. These modifications do not change the control functions or alarms.

The modifications do not result in an increased probability of a rupture of a volume control tank, nor do they alter the VCT rupture analysis found in FSAR Section 14.2.3. There is no net additional load on the inverters or on the station batteries. Existing instrument bus separation is maintained. (SER 90-112-01)

24. MR 90-272 (Unit 1), 480 V Electrical System. MR 90-272 changes the existing control circuits for bus tie breaker 1B52-15C (1B01 to 1B03) [1B52-18C (1B02-1B04)] and bus supply breaker 1B52-4B (1B01) [1B552-5B(1B02)] to allow reclassification of cables ZA1B15CC and ZA1B15CD [ZB1B18CC and ZB1B18CD] from safety-related to non-safety-related. This corrects electrical separation conflicts found in the control circuits for bus tie breakers 1B52-15C (Train "A") and 1B52-18C (Train "B").

Summary of Safety Evaluation: The modification corrects the electrical separation conflicts via two isolation relays. Since the relays are located in the safety-related switchgear section which houses the tie breakers, use of the isolation relays does not result in an electrical separation concern.

The control circuits for bus tie breakers 1B52-15C (train "A") and 1B52-18C (train "B") have adequate electrical separation. The modification does make changes to the control circuits for these breakers, but does not change the intended function of the affected portions of the control circuits. The control circuits remain functionally the same.

The safety-related function of these bus tie breakers is to trip upon either bus undervoltage or safety injection. MR 90-272 does not alter this function of the control circuits. The probability of an accident or equipment failure is not increased. (SER 91-089-01)

The following modifications were installed in 1992:

1. MR 84-227*A (Unit 1), MR 84-228*B (Unit 2), IWPs 84-227*A-1, 84-227*A-2, 84*228*B-1, 84-228*B-2. Inverters. The MRs replace the existing inverters 1DY01, 1DY02, 2DY01, and 2DY02 with new inverters. The new inverters are QA, safety-related, seismically and environmentally qualified.

Summary of Safety Evaluation: During the inverter replacement, the appropriate red or blue instrument bus is fed from its swing inverter (DY0A or DY0B) and does not affect the installation.

The new inverters were seismically and environmentally qualified, and are seismically mounted. The new inverters are mounted in the same location, thus maintaining safety train separation. (SER 91-083-01)

2. MR 87-016*E (Common), Fire Protection System. A new fire detection panel (C933) was added to the simulator to enhance protection of its components.

Summary of Safety Evaluation: The C933 panel does not monitor safe shutdown equipment areas, nor does it impede control room alarms from safe shutdown fire detection panels. There are no Appendix R-related design concerns associated with this

portion of the fire detection system. The C933 simulator panel design and its tie to the D414 NSB panel are consistent with the design of existing plant fire detection and alarm systems. The tie-in is not QA-Scope based on FPER Section 2.6.3. (SER 91-106-01)

3. MR 87-156*A, MR 87-156*B, MR M-784 (Common), TM 87-039, HVAC.
MR 87-156*A installs a 6 kW electric duct heater in the ventilation supply duct of PAB battery room A (D105) on PAB El. 26'. An environment in the battery room is required that allows the proper float voltage on the battery to be maintained without affecting the electrical equipment room temperatures. The MR maintains a temperature of 77±5°F in the battery room and removes TM 87-039 to allow proper air flow through the duct. The MR also removes a wall-mounted space heater.

Summary of Safety Evaluation: The D105 battery and electrical equipment operation does not affect installation of the modification. The modification installation has no impact on safety-related equipment. During the duct heater installation, the interim conditions compromise the integrity of the ventilation system of battery room A by cutting a hole in the side of the duct. However, calculation 13754.16 E(N)-2 documents hydrogen produced by the battery does not reach combustible concentrations until the ventilation system has been completely inoperable for 11 days. The maximum time the hole exists in the duct is 4 hours. The ventilation system continues to operate during that time so the battery room is not completely isolated from air circulation. The battery life and reliability improved by allowing the proper float voltage to be maintained. The electrical equipment rooms that serve the same ventilation system do not affect the installation of the heater nor the final configuration of the modification.

The heater installation does not adversely affect the seismic integrity of the battery room ventilation system. Calculation 6904-34-HS documents the duct and existing supports to withstand the additional weight of the heater and maintains their Seismic Class 1 rating. The calculation also verifies the method of attaching the heater to the duct meets Seismic Class 1 criteria. All materials used are compatible with existing duct material to avoid potential corrosive effects from incompatible materials.

The modification does not add to the fire loading in the battery room because the heater is located in the air inlet ducts upstream of the fire damper. The existing heater on the north wall of the battery room is removed following installation and testing of the new duct heater. All penetrations into the battery rooms from the old heater are sealed.

Electrical power for the heater and the associated controls are supplied from non-safeguards MCC 2B41. The duct heater adds a total of 7.2 amps continuous to 2B41. Calculation N-92-007 load justification, was performed to verify the acceptability of adding the heater load to the MCC. There are no overloads or other adverse affects on the non-safeguards electrical distribution system. The safeguards power equipment, including emergency diesels are unaffected by the modification during installation or following completion. During a loss of offsite electrical power event, MCC 2B41 loses power and deenergizes the heater, which causes the battery room to be cooled. (SER 92-013) and (SER 92-013-01)

4. MR 88-091 (Common), Pipe Penetrations. MR 88-091 installs a radiation shield box for pipe penetrations in the spent fuel pool demineralizer cubicle wall to reduce radiation levels because of the piping.

Summary of Safety Evaluation: A shield box was fabricated from angle iron and 1/4" thick carbon steel plates. The box fits around the pipe penetrations and pipe elbows, to cover them. Steel provides a minimum 3" thickness to reduce the high radiation levels.

In addition, the box is Seismic Class 1 because it is attached to a Seismic Class 1 wall and covers Seismic Class 1 pipes. (SER 91-111)

5. MR 88-0971A, (Unit 1), HHSI, CS and RHR Full Flow Test Lines. The MR installs new full-flow test lines in the Unit 1 high head safety injection (HHSI), containment spray (CS), and residual heat removal (RHR) systems.

Summary of Safety Evaluation: The new test lines were installed in response to NRC Bulletin 88-04 as refined by NRC Generic Letter 89-04, to allow for an increased flow rate through the pumps during inservice testing. The flow rate is increased to protect the pumps from the adverse effects of hydraulic instability at low flow rates. The test lines are isolated from their respective systems during normal operation. The isolation is to normally shut gate valves or if the seat leakage through the gate valves exceeds the acceptance criteria of the Leakage Reduction and Preventive Maintenance Program (NUREG-0578), a blank flange installed in place of the orifice plate in the flow detector. The response concluded since the majority of pump operations associated with miniflow conditions is related to the inservice testing program, installation of higher capacity test lines significantly reduces cumulative pump operation with less than recommended flow rates. The inservice test flow rates are increased to 700 gpm, 1200 gpm, and 1560 gpm for the HHSI, CS and RHR pumps respectively, as specified in FSAR Section 6.

The new test line for the SI system is routed from the pump discharge cross-connect line between 1SI-829A&B through the flow orifice, a flow control valve (1SI-829C), a manual gate valve (1SI-829D) to a return line shared with the CS system back to the RWST. For the CS system, a new pump discharge cross-connect line is installed with two manual gate valves (1SI-862G&H). The test line is routed from the between 1SI-862G and H through a flow orifice, a flow control valve (1SI-862J), a manual gate valve (1SI-862K) to a common return line back to the RWST. For the RHR system, there is an existing section of pipe between the "B" train heat exchanger discharge piping and 1SI-742, return to RWST isolation valve. The new test line replaces this section of pipe with a piping complex which includes an isolation valve, a flow measuring orifice and a flow control butterfly valve. The piping complex also ties into the "A" train heat exchanger discharge piping.

The modification adds a vent line to the high point of RHR "A" train as corrective action to Condition Report 91-132. The condition report identified a problem with air being trapped in 1HX-11A during inservice testing. The vent is installed in RHR "A" train to CS pump suction line to allow for better venting of the RHR train.

FSAR Section 6.2 states the piping and supports are designed for 1745 psig and 300°F. Calculation N-90-067 identifies the maximum operating temperature to be 206°F in the HHSI pump discharge cross connect piping. The piping and supports for the modification are designed for 1745 psig and 210°F in the pump discharge cross connect piping and 1745 psig and 100°F for the remainder of the test line. This is the expected temperature distribution with the test line isolated and the HHSI system in the recirculation mode. This is the expected worst case thermal condition and is more accurate than analyzing the new test line at 1745 psig and 300°F.

The accidents in FSAR Sections 6.2 and 6.4 that involve the HHSI, RHR and/or the CS systems are the steam generator tube rupture, steam line rupture, and loss of coolant accident (LOCA). The three systems act as a post-accident mitigating system. The HHSI and RHR systems are connected to the RCS which presents the possibility of an intersystem LOCA. The new piping for the HHSI and RHR systems is outside the boundaries of the containment isolation valves and during normal system operation is isolated from the respective systems by two isolation valves. The construction and

integrity meet the same specifications as the current HHSI, and RHR systems. There is little possibility of a LOCA via the new test lines.

During a LOCA, the potentially high radioactive primary coolant is not circulated through the HHSI, CS or RHR systems until the systems are lined up for containment sump recirculation, which requires operator action outside the control room. If a LOCA occurred while conducting a full flow test on the CS, HHSI or RHR systems, the test operators have time to isolate the flow path to the RWST. (SER 91-074-08)

6. MR 88-097*B, IWP 88-097*B, (Unit 1), HHSI, CS and RHR Full Flow Test Lines. The MR design package controls the installation of piping, valves, flow orifices and pipe supports for MR 88-097 prior to the U1R19 outage. Portions of the piping and selected valves are installed prior to the U1R19 outage to reduce the time needed to complete the modification during the outage. The piping for the safety injection (SI) and containment spray (CS) test lines is installed in the SI and CS pump room. A 10" hole is bored in the south wall of the pump room to allow for routing of the test line to the RWST.

   Summary of Safety Evaluation: The piping for the residual heat removal (RHR) system is installed in pipeway No. 2 valve gallery with installation to be conducted with both units operating at power. The SI, CS and RHR systems are required to be in operation per TS 15.3.3. Where possible, the piping is installed using the analyzed supports. When needed, temporary supports are used to secure components in place. These temporary pipe supports are interim installation configurations and were replaced with permanent supports under MR 88-097*A during U1R19 outage.

   Construction is in accordance with B31.1-1967. A main concern during installation is the protection of the operating SI, CS and RHR systems. The systems are protected by ensuring all Seismic 2/1 criteria are met and by following the appropriate administrative controls for transient combustibles, scaffolding, penetrating fire barriers and ignition controls.

   The 10" hole requirement bored in the south wall of the SI and CS pump room requires a maximum of two reinforcing bars in each the horizontal and vertical directions may be cut without reducing the rating of the wall. This is verified by calculation 100447-050-C-038. (SER 91-074-09)

7. MR 88-097*D, (Unit 1), SI, RHR, and CS Test Line Flow Transmitters. The design package adds local reading flow transmitters to the safety injection (SI), residual heat removal (RHR) and containment spray (CS) test lines. These flow transmitters are required to obtain pump flow data during ASME Section XI testing.

   Summary of Safety Evaluation: Installation of the transmitters occurs when the SI, RHR, and CS systems are out of service for installation of the new test line piping. The transmitters and sensing lines are seismically mounted to provide assurance against primary system leakage if a LOCA and a seismic event occurred during test line use. The transmitters and the sensing lines represent a closed system which contains primary coolant should the test line isolation valves leak.

   The transmitters do not perform a safety-related function and are powered from local lighting panel circuits. All conduit is installed Seismic Class 1 to avoid potential impact with safety grade equipment located in the surrounding areas. The transmitters, tubing, isolation valves, and fittings exceed the requirements for the 1501R SI pipe class, and are adequate for use as proposed.

   The transmitters meet the accuracy requirements of Appendix C of the Inservice Testing Program. (SER 91-074-12)

8. MR 88-097*E, (Unit 1), Safety Injection System. The modification replaces the lower rings of packing for valves 1SI-868A&B and replaces it with a stainless steel spacer. The upper rings of packing, above the lantern ring, serve as the stem packing. In addition, the existing 1/2" leakoff connection is configured to function as a test line with a 1/2" valve and plug.

Summary of Safety Evaluation: The plug is removed for valve testing. The valves are modified to allow leak rate testing to be performed using the leakoff line as a pressurization point to pressurize the valve body between the discs. This is performed to allow Appendix J testing to be performed in the proper direction. Appendix J testing is required to allow the valves to be used as containment isolation valve during containment spray system pump performance testing.

Modification of the leakoff line to accept a test line does not affect the capability of the valve to fulfill its requirements. The leakoff line forms the pressure boundary for the valve when the valve is open, and requires the weld used to connect the line to the test line fittings be pressure tested in accordance with a pneumatic test. This is followed by the performance of refueling outage leak testing. The leakoff line plug integrity may be tested during the performance of annual inservice leak testing.

This modification does not impact on the ability of the valves to perform their required function. This modification is installed during plant cold shutdown conditions when containment integrity is not required so valve disassembly does not affect plant operability. (SER 91-074-10)

9. MR 88-099*D, (Unit 1), Auxiliary Feedwater System. MR 88-099*D replaces the existing conduit supports for P38A&B auxiliary feedwater pumps. The new conduit supports to provide additional space needed for the installation, operation, and maintenance of new control valves on the auxiliary feedwater recirculation line.

Summary of Safety Evaluation: The affected conduit supports and conduit for P38A&B are classified as seismic and safety-related. The existing conduit supports for each train, remain in service until the installation of the new conduit supports required for a given train, or the installation of temporary conduit supports for a given train is complete. After installation, the old conduit supports for the train are removed. Operability of pumps P38A&B are not affected during installation. (SER 91-025-01)

10. MR 88-114 (Common), Control Room HVAC. MR 88-114 replaces C67 with a new panel containing a different pattern for the switches.

Summary of Safety Evaluation: The new panel is seismically mounted in the same location per calculation N-91-133. The angle frame beneath the existing panel is not needed. The panel was seismically evaluated via evaluation work sheet G-20.

During installation, the HVAC system is in Mode 3 with 100% recirculated air, 25% of which is recirculated through the charcoal filters. While in Mode 3, the plant is not in a Limiting Condition of Operation (LCO). During some periods of time, both fans of a redundant pair are off at the same time. During these periods, the plant is in an LCO per TS 15.3.12. These periods of time are minimized and are not allowed to total more than 7 days.

While in Mode 3, the computer room is isolated from normal air conditioning and may not receive adequate cooling. Fans are used and doors opened, as required, to cool the computer room. Appropriate compensatory measures for security and fire protection are provided.

If necessary for computer room cooling, it is permissible to open ventilation dampers to the computer room. When these dampers are open, the system is no longer in Mode 3 and is in an LCO per TS 15.3.12. The total time in this configuration when added to the time when both fans of a pair are off shall be limited to less than 7 days. (SER 91-025-01)

11. MR 88-130 (Common), Computer System. MR 88-130 provides a component cooling (CC) water surge tank level signal to the PPCS for both units. Computer isolation blocks are inserted in existing CC surge tank level instrument loops, 1L618 and 2L618. The additional inputs to the PPCS to provide trending data to aid in problem diagnosis.

Summary of Safety Evaluation: The CC surge tank level indication/alarm is used in identifying leaks described in the FSAR. A high level indication/alarm may indicate a large tube side to shell side leak in a residual heat exchanger (FSAR Section 6.5-8). A high level indication/alarm may also indicate a thermal barrier cooling coil rupture in a reactor coolant pump (FSAR Section 6.5-9). A low level indication/alarm indicates a leak in the CC system, such as severance of the cooling line serving a reactor coolant pump cooler (FSAR Section 9.3-15).

Since the reliability of the instruments is not affected, the indication and alarms still perform as before. The operator has the additional capability of trending the level on the PPCS to help identify the problem. This may enhance the diagnosis and minimize the consequences of an accident.

This change requires installing #16 AWG TSP between the computer MUX in the computer room and the instrument racks in the control room. The additional cable does not overload any raceways beyond their 40% fill capacity and does not present a significant combustible load increase. (SER 90-022)

12. MR 88-188*D, IWP 88-188*D-1 Through IWP 88-188*D-26. Miscellaneous Motor-Operated Valves (MOVs). The modification replaces 2-rotor limit switches with 4-rotor limit switches for 15 MOVs, and rewires position indication at the motor control center to provide indication thermal overload device actuation for 11 valves.

Summary of Safety Evaluation: The limit switch replacement was performed on the following valves in Unit 2: SI-826A, SI-851A, SI-851B, SI-852A, SI-852B, SI-856A, SI-856B, SI-860B, SI-871A, SI-871B, RH-700, RH-720, CV-112B, MS-2020 and SW-2907. This changeout enhances the accuracy of valve status information in the control room by providing independent limit switch rotors for two functions previously wired off a single rotor: shut valve position indication and open torque switch bypass. Since each rotor corresponds to one valve-stem position, splitting these two functions to independent rotors allows for separate limit switch setpoints for the shut valve position indication and the open torque switch bypass. This replacement also enhances valve operability by allowing the open torque switch to be bypassed longer and thus reducing the possibility of the motor tripping out on open torque and the valve not opening.

The valve position indication rewiring that provides detection of motor thermal overload actuation was performed on the motor control centers for the following valves in Unit 2: SW-2880, SW-2907, SW-2908, CW-0001, CW-0002, CW-0003, CW-3501, CW-3502, CS-2189, CS-2190 and AF-4006. This wiring change provides the control room with indication of valve inoperability due to motor thermal overload actuation by having the associated valve position indication lights deenergize when the overload blows. Currently, the valve position indication lights remain energized and the control room does not have indication that the valve was inoperable.

The new limit switch assemblies are procured as EQ equipment for EQ valve operators and EQ wire were used. The additional rotors add negligible weight to the operator and therefore, do not affect the seismic analysis.

The modifications are installed during a refueling outage and/or in accordance with Technical Specification requirements. (SER 92-066)

13. MR 88-188*D, IWP 88-188*D-27, SER 92-066. Miscellaneous Motor-Operated Valves (MOVs). SER 92-066 was previously approved for installation of 4-rotor limit switches on various motor operated valves during U2R18. Subsequent to approval, it was determined that one additional valve, 2CV-285 excess letdown heat exchanger discharge isolation valve should be modified by adding a 4-rotor limit switch.

Summary of Safety Evaluation: The 2CV-285 operator is a QA component and is not environmentally qualified.

Four-rotor installation is performed when the valve operator is removed for rebuilding per MWR 913885. Interim conditions for 2CV-285 are that excess letdown be out of service during return to service testing of the valve. Work should be performed during 2CV-285 maintenance work, with Unit 2 in cold shutdown for containment entry. The valve is electrically inoperable but may be manually repositioned if required.

New 4-rotor switch assemblies are mounted in a similar fashion to the existing 2-rotor assemblies and are of comparable weight. The additional rotors add negligible weight to the operator and does not impact the seismic qualification based on engineering judgement. (SER 92-066-01)

14. MR 88-188*D, IWP 88-188*D-13, Miscellaneous Motor-Operated Valves (MOVs). The original SER approved this work when residual heat removal (RHR) would be out of service. Outage scheduling constraints require the limit switch upgrade be performed with RHR in service. Post-modification testing is performed with RHR out of service.

Summary of Safety Evaluation: Plant outage scheduling required the limit switch be replaced on 2RH-700 when the valve is open and RHR is in service. 2RH-700 remains open until steam generators are available for decay heat removal and RHR can be removed from service. 2RH-700 is then cycled for post-modification testing.

The plant conditions specified in the installation work plan include the following: 2RH-701 is operable from the control room; the reactor coolant system (RCS) is filled and vented, prior to 2RH-700 power tagout; 2RH-700 remains open during this work and is only shut (manually) if required to isolate an RCS leak, or if RHR can be removed from service (SGs assume DHR load); and an assigned operator is appointed to manually shut 2RH-700, if required. (SER 92-066-02)

15. MR 88-188*D, IWP 88-188*D-28 and D-29, SER 92-066, Miscellaneous Motor-Operated Valves (MOVs). SER 92-066 was approved for installation of 4-rotor limit switches on various motor-operated valves during U2R18. Subsequent to approval it was determined that additional valves, 2CC-738A&B, residual heat removal (RHR) heat exchanger component cooling inlet valves, should be modified.

Summary of Safety Evaluation: The valves are electrically inoperable during this work but may be manually opened as necessary to maintain their function during accident conditions. Work controls specify one of the 2CC-738A or 738B valves be worked at a time, and an assigned operator be provided to manually open the valve if needed. Position indication is not available in the control room after power is removed for limit switch changeout. Valve position can be determined locally by examining valve stem

position. Backup indication is provided by means of observing RHR heat exchanger outlet temperature in the control room.

The 2CC-738A&B operators are QA, environmentally qualified, safety-related, Seismic 1 components. EQ wire is used to maintain operator EQ requirements. QA design controls and components are utilized. (SER 92-066-03)

16. MR 88-188*F, IWP 88-188*F-1 Through IWP 88-188*F-26, (Common), Miscellaneous Motor-Operated Valves (MOVs). The modification replaces limit switches on Unit 1 valves: SI-825A, SI-825B, SI-851B, SI-852B, SI-856B, SI-860B, SI-860C, SI-871A, SI-871B, CC-738A, CC-738B, CV-112B, MS-2020 and CV-270B.

The valve position indication rewiring that provides detection of motor thermal overload actuation is performed on the motor control centers for the following valves in Unit 1: SW-2880, SW-2907, SW-2908, CW-0001, CW-0002, CW-0003, CW-3501, CW-3502, CS-2189, CS-2190 and AF-4006. The wiring change provides the control room with indication of valve inoperability due to motor thermal overload actuation by having the associated valve position indication lights deenergize when the overload blows.

Summary of Safety Evaluation: The design package requires the valve being worked on be isolated and inoperable for about 12 hours. The valves are worked one at a time. The exact system isolation boundaries and the position of the valve being worked on is determined at the time of installation consistent with any other current work performed. (SER 90-039-02)

17. MR 89-014 (Common), Main Control Boards. MR 89-014 replaces the existing annunciator tiles on the main control boards and auxiliary safety instrumentation panels (ASIPs). The new tiles are installed to resolve human engineering discrepancies (HEDs) documented during the control room design review.

Summary of Safety Evaluation: The main control boards, annunciator panels, and alarms are mentioned in the FSAR. Replacement of the annunciator tiles changes the facility as described in the FSAR. There are no changes in the function or operability of the annunciators resulting from the tile replacement. (SER 90-006)

18. MR 89-026*A (Unit 2), Feedwater System. The modification replaces the existing fixed orifice in the fourth pass drain line from the moisture separator reheaters (MSRs) with a throttle valve. This allows the reheat steam flow through the MSR to be more efficiently controlled.

Summary of Safety Evaluation: A failure of this line creates similar conditions to a steam line break and/or a loss of feedwater enthalpy accident. The accident analyses however, assume failures of a much larger magnitude. Therefore, this change is not outside of the existing safety analysis.

The HELB accident addresses steam line failures in the general area of the proposed change. The analysis assumes failures of a larger magnitude than possible with the proposed change, and was determined that safe-shutdown capabilities is not affected.

Valve mispositioning during operation does not create potential nuclear safety concerns. The effect of a mispositioned valve would be localized to the parent system. The status of valve position is checked periodically by monitoring system temperatures to assure proper positioning. (SER 92-077)

19. MR 89-118 (Common). Waste Gas System. MR 89-118 installs a sight glass flow detector and a 3/4" tee, valve and capped nipple on the gas decay tank (GDT) common drain line.

Summary of Safety Evaluation: Failure of this installation has no impact on the FSAR accident analyses or other postulated accidents because this line is not associated with safety-related equipment or equipment which could be required to mitigate the consequences of an accident. FSAR Section 11 however, contains a statement of evaluation of a failure (leak) of a GDT. The modification could have the consequences described in that evaluation if it failed; however, this is extremely unlikely for several reasons: 1) There is a normally shut isolation valve between each GDT and this common drain line. Each isolation valve is infrequently operated and is physically very close to the modification installations. They are manual valves; therefore, an operator is able to isolate the valve(s) if the modification failed with an isolation valve open; and 2) The design is in accordance with B31.1 and the original design criteria for the system.

During installation, the GDT common drain line is opened just downstream of the four tanks' first-off isolation valve. This temporarily changes the configuration as described in the FSAR. This configuration does not adversely affect the possibility of a malfunction because the first-off valves are administratively controlled and are in the shut position. The first-off valve diaphragms are changed out prior to installation and the GDTs are depressurized. (SER 90-016)

20. MR 89-121 (Unit 2). Safety Injection (SI) System. The modification replaces the 25 ft-lb motor on MOVs 2SI-878A, 2SI-878B, and 2SI-878C.

Summary of Safety Evaluation: Replacement of the valve operator motors with smaller torque motors does not affect the safety injection (SI) system. The motor replacement does not impair the ability of the valve operator from performing its function or increase the occurrence of malfunction.

The lower torque motor (15 ft-lb) is the correct size for the valve operators and the valves (2SI-878A, 2SI-878B, and 2SI-878C). The valves operate as they have before, but with less likelihood of valve damage due to stem over-thrust. Calculation P-90-017 verified the stem thrust capacities for both the 15 and 25 ft-lb motors. The valves operate as originally intended; the 15 ft-lb motors do not change the function, reaction time, or other valve characteristics important to safety. During motor changeout, the valve can be manually operated if required. The valves are positioned to the SI train available if required, as an alternate boration path or in cases of reduced inventory within the RCS. The valves are stroke tested after installation, but timing of the step is planned to ensure at least one other train is available or that the SI system is not required to perform any functions important to safety. The loss of one SI train was previously evaluated but does not normally apply to refueling shutdown conditions.

The SI system is still able to perform its function during and after motor changeout. Replacement of the motors does not initiate an accident or cause any equipment to malfunction. The valve and SI system is still operable during motor replacement. (SER 92-084)

21. MR 89-188 (Common). 125 V DC System. MR 89-188 replaces DC breakers in distribution panels D11 and D13 with breakers having a thermal and magnetic trip feature.

Summary of Safety Evaluation: The original evaluation supplied both unit's A01, A02 B01 and B02 DC control power from D11. The addition of the magnetic trip element prevents non-safety-related cable faults from exceeding the D06 battery duty cycle ratings, and from reducing battery capacity.

During the brief period required to transfer the Unit 1 A01 and A02 DC control power from D11 to D13, the following protective relaying features for Unit 1 are lost: reactor coolant pump (RCP) breaker open on undervoltage (UV) and underfrequency (UF) to the reactor trip circuit; AMSAC start circuitry for the motor and turbine-driven auxiliary feedwater pumps do not start on main feed pump breaker opening; 4 kV UV initiation of the turbine-driven auxiliary feedwater pump; and 4 kV UV and UF trips for 1A01 and 1A02 (non-safety-related).

All other reactor trip signals and auxiliary feedwater signals remain operable.

The duration of the transfer is expected to be within 5-10 seconds. During switching of control power for 1B01 and 1B02, UV load shedding and remote indication is lost. This is not a safety concern.

The 2B01 and 2B02 breakers continue to provide overcurrent protection for attached loads; however, the electrical close and open functions, along with breaker remote position indication are lost. UV protection for 2B01 and 2B02 is supplied from 2B03 and 2B04, respectively. 2B01/2B02 bus tie breakers automatically open on 2B03/2B04 (respective) UV condition. 2B01 and 2B02 do not automatically resequence. This is not a safety concern.

After 2B01 and 2B02 are properly configured, the supply breakers for 2A01 and 2A02 are opened to downpower the buses. The D11 and D13 DC breakers for 2A01 and 2A02 are also opened until suitable 70 amp thermal/magnetic replacement breakers can be located. 2A01 and 2A02 are not needed while Unit 2 is in cold shutdown. (SER 89-134-01)

22. MR 89-188*E (Common), 125 V DC System. MR 89-188*E replaces breaker D11-25 with a Type HFA breaker and removes breaker D11-26.

Summary of Safety Evaluation: A single fault associated with the DC distribution does not prevent automatic feed isolation since the main feed regulating valves shut in such a scenario.

The single fault does not prevent reactor trip on 2A01/2A02 underfrequency to reactor coolant pump trip above P8. Below P8 reactor trip is dependent upon low flow trip. This is no different than the current case of loss of DC control power to a single non-vital 4 kV bus.

Steam-driven auxiliary feed pump initiation of opening of both main feed pump breakers on UV is lost. The motor-driven auxiliary feed pumps provide a backup. (SER 89-134-05)

23. MR 90-014*A (Unit 2), Main Control Boards. MR 90-014*A changes the contacts on the motor-operated valves that supply the safety injection (SI) spray ready status panel indications.

Summary of Safety Evaluation: These valves had the two-rotor limit switches replaced by a new four-rotor limit switch per MR 88-188*D. This aids the operators by giving them correct indications on the SI spray ready status panel and provides better assurance of proper valve positioning of the associated systems.

Work was performed during the U2R18 refueling outage, safety injection and containment spray systems are not required. (SER 92-068)

24. MR 90-034 (Unit 1), Motor Operated Valves (MOVs). The modification replaces existing SMB-000 motor operators with larger SMB-00 operators for three motor-operated valves in Unit 1: PORV block valves, RC-515 and RC-516, and charging line isolation valve, CV-1298.

Summary of Safety Evaluation: The increased thrust provided by the larger SMB-00 motor operator provides additional margin to ensure the associated valve shuts under maximum flow or opens under maximum differential pressure conditions, even at reduced supply voltage. The 4-rotor limit switches enhance the accuracy of valve shut position indication in the control room and improves valve operability by providing independent limit-switch rotors for two functions previously wired off of a single rotor: shut valve position indication and open torque-switch bypass. Since each rotor corresponds to one valve-stem position, splitting these two functions to independent rotors allows for the shut valve-position indication to correspond to actual valve seating, and allows the open torque switch to be bypassed longer, thereby reducing the possibility of the motor tripping out on open torque.

The modification requires the valve be isolated and inoperable for about 30 hours. Testing requires system isolation for approximately 2 hours to allow valve cycling. The exact system isolation boundaries are determined at the time of installation consistent with other current work performed.

The new motor operators are procured as EQ equipment for EQ valve operators and EQ wire is used for installation on EQ valves. Seismic considerations were addressed in the modification design package. The existing seismic support/structures were determined to be adequate for the larger operators.

The modification is installed during a refueling outage in accordance with Technical Specifications. Once installed, the larger motor operators provide improved valve operability (increase thrust and longer open torque-switch bypass time). (SER 92-027)

25. MR 90-035 (Unit 2), Motor Operated Valves (MOVs). The modification replaces existing SMB-000 motor operators with larger SMB-00 operators for three motor-operated valves in Unit 1: PORV block valves, RC-515 and RC-516, and charging line isolation valve, CV-1298.

Summary of Safety Evaluation: The increased thrust provided by the larger SMB-00 motor operator provides additional margin to ensure the associated valve shuts under maximum flow or opens under maximum differential pressure conditions, even at reduced supply voltage. The 4-rotor limit switches enhance the accuracy of valve shut position indication in the control room and improves valve operability by providing independent limit-switch rotors for two functions previously wired off of a single rotor: shut valve position indication and open torque-switch bypass. Since each rotor corresponds to one valve-stem position, splitting these two functions to independent rotors allows for the shut valve-position indication to correspond to actual valve seating, and allows the open torque switch to be bypassed longer, thereby reducing the possibility of the motor tripping out on open torque.

The modification requires the valve be isolated and inoperable for about 30 hours. Testing requires system isolation for approximately 2 hours to allow valve cycling. The exact system isolation boundaries are determined at the time of installation consistent with other current work performed.

The new motor operators are procured as EQ equipment for EQ valve operators and EQ wire is used for installation on EQ valves. Seismic considerations were addressed in the modification design package. The existing seismic support/structures were determined to be adequate for the larger operators.

The modification is installed during a refueling outage in accordance with Technical Specifications. Once installed, the larger motor operators provide improved valve operability (increase thrust and longer open torque-switch bypass time). (SER 92-067)

26. MR 90-064*A (Common), Blowdown Evaporator (BDE). The design package installs a new section of piping into the BDE distillate line that permits replacement of the existing conductivity cell. The new piping also permits the conductivity cell to be isolated from the BDE system prior to removal for cleaning or replacement.

Summary of Safety Evaluation: The new bypass line is installed during a BDE outage and temporarily prevents liquid waste processing. The defective conductivity cell was removed from service and daily chemistry sampling is performed until it is returned to service. The replacement conductivity cell and meter is installed after completion of the piping installation. Operating procedures were revised as necessary to control operation of the new BDE system configuration and conductivity cell removal.

The modification was designed in accordance with the original installation code and applicable QA requirements for the blowdown evaporator system. The pressure retaining ability of the system is not degraded. (SER 92-002)

27. MR 90-064*B (Common), STPT 16.1, Blowdown Evaporator (BDE). The design package replaces the blowdown evaporator conductivity cell and revises the setpoints.

Summary of Safety Evaluation: STPT 16.1 was reviewed to determine the alarm and valve control setpoints for the new equipment. From this review, it was discovered these were original setpoints, based on reusing the water. Since the flow path to the condensate storage tanks was eliminated by MR 90-134*B, high quality water is not required. Therefore, the BDE conductivity high alarm setpoint was changed from 100 μmho to 50 μmho and the BDE conductivity setpoint for valve control from 5 μmho to 50 μmho.

MR 90-064*B and STPT 16.1 do not directly affect nuclear safety. The conductivity cell temperature and pressure specifications exceed the design ratings of the blowdown evaporator system. The distillate tanks are sampled prior to and are monitored for radiation during discharge. (SER 92-072)

28. MRs 90-093 (Unit 1) 90-094 (Unit 2), Instrumentation. The modifications replace the local indicators and flow switches which provide component cooling (CC) water low flow alarms for the residual heat removal (RHR), safety injection (SI) and containment spray (CS) pumps.

Summary of Safety Evaluation: The new instrumentation is programmed to alarm at the same setpoint as the existing switches. The transmitters and digital indicator are located near safeguards equipment on PAB El. 8'. The transmitters and indicators are seismically mounted.

The new instrumentation is powered from the control room via spare breakers from non-safeguards instrument buses 1Y06 and 2Y06. These buses are capable of being fed by the emergency diesel generators so the alarms function following a loss of AC accident. The added load on the buses and diesels is negligible. If one of the instrument buses fail, the component cooling water low flow alarms do not function.

Failure of the alarms does not affect CC flow as the alarms are independent of any control circuits. Other CC flow indication is still available to indirectly monitor flow via monitoring pump and inventory parameters such as CC water pump discharge low pressure and CC surge tank level.

Spare breakers are created by combining both Unit 1 RHR cubicle drain valves on one breaker supplied by 1Y06 and both Unit 2 valves on one breaker supplied by 2Y06. Each RHR cubicle drain valve is supplied by a separate 10 amp breaker. A single 10 amp breaker is capable of supplying both loads as each valve circuit only draws ~0.5 amps. Combining the valves does not result in functional changes. The valves are not redundant or safety-related.

Following installation, the transmitters and digital indicators are calibrated to ensure they are functional and within tolerance. The control room alarms are tested as part of the calibration. The modification is QA-Scope because the pressure retaining ability of the new transmitters. Prior to acceptance, a pre-calibration pressure test and an in-service leak check is performed. (SER 92-003)

29. MR 90-105 (Unit 1), Chemical and Volume Control System. MR 90-105 provides a common hard pipe vent path from hydrogen regulators 1CV-113 and 1CV-158, and nitrogen regulator, 1CV-114 to the volume control tank (VCT) cubicle exhaust duct.

Summary of Safety Evaluation: There is no change in the gases being vented; amount, type or radiological activity. Appendix A lists portions of the chemical and volume control system (CVCS) as Seismic Class 1 and the entire auxiliary building ventilation system as Seismic Class 1.

Existing CVCS valves can be used to isolate the leak and nearby existing CVCS connections can be used to vent gas downstream of any of the regulators to the VCT valve gallery, which is the current accepted practice. A vent line failure resulting in the duct tie-in plate pulling away from the ductwork gives an opening in the ductwork of less than 1 square inch, a negligible amount when compared to the nearby intake grille area of 216 square inches. The failure of the vent line does not adversely affect any system, structure, or component important to safety.

The double isolation valves from each CVCS tie-in meet the waste gas system design requirements. There is no increase in the discharge of radioactive gases. The design and layout has a negligible effect on the forces transmitted to the CVCS and the auxiliary building ventilation system. Seismic 2/1 concerns are satisfied. There are no adverse system interactions. (SER 91-076-01)

30. MR 90-120 (Common), Doors. MR 90-120 replaces four fire/security doors to facilitate movement of carts and equipment. Three of the doors are fire doors. The new fire doors are UL approved and meet the requirements of the Fire Protection Evaluation Report.

Summary of Safety Evaluation: The new doors are equal to or better than the doors replaced. The new doors are of heavier construction. Installation of the new doors requires a fire barrier penetration permit, and a security officer to be present when the door is removed. Previous evaluations for fire protection and security are not affected. Safe shutdown area fire doors remain painted as indicated in our response to NRC IR 82-17. (SER 91-066)

31. MR 90-134*B (Common), Piping Reroute. The modification reroutes plant heating system condensate return lines, (6"-JB-14 and 2"-JB-13), relocates a support on line 6"-JG-3, and removes a portion of the waste evaporator distillate line 1-1/2"-BED-152.

Summary of Safety Evaluation: The change to the plant heating system condensate return involves rerouting the 6"-JB-14 and 2"-JB-13 lines. The line is a continuation of the plant heating system condensate return to the auxiliary feedwater (AFW) system. The line was seismically reanalyzed and the boundary relocated adjacent to the functional "Q" boundary of the AFW system at valves AF-10 and AF-69. The change does not affect the description, function, or operation of the AFW system, as described in the FSAR.

Demineralized water to the auxiliary feedwater system, line 6"-JG-3, requires relocation of a deadweight support to avoid interference with the new wall. The line is non-seismic and non-safety related. The support is required for deadweight consideration only.

The change to the waste disposal system involves removing a portion of line 1-1/2"-BED-152 from the waste distillate pumps to the condensate storage tanks. This portion of the line was not used since initial operation. This changes does not adversely affect the function or operation of the system. (SER 92-004-01)

32. MR 90-134*C (Common), 125 V DC System. The modification involves the civil/structural/architectural work associated with the addition of new 125 V DC Class 1E and Non-Class 1E batteries, in the Class 1 portion of the turbine building at El. 26'. The civil/structural work involves the addition of new reinforced concrete walls and slabs, including embedded steel, and the addition of miscellaneous steel stairs, platforms, ladders, and handrail. The architectural work involves the replacement of CMU walls, the replacement and/or addition of new hollow metal doors, frames and hardware, and field-applied coatings for floors and walls.

Summary of Safety Evaluation: The modification adds additional structures to the floor at El. 26' of the Class 1 portion of the turbine building, thereby increasing the mass of the floor and the load on the existing slab, supporting beams, walls, and foundations. The Class 1 portions of the new structures are designed for severe environmental loads of earthquake and tornado loads in addition to dead and live loads. The existing structures were checked for the new loads imposed by the new structures and equipment.

The existing structures were analyzed for the additional loads imposed upon them by the new structures and equipment. The new structures enclosing and protecting Class 1 equipment were designed for both the design and the maximum hypothetical earthquake and for tornado, wind and missile loads. Due to venting, tornado pressure loads are not applicable. Resulting stresses in the new and existing structures are within design allowables.

The existing structures were checked for the new loads imposed by the new cubicles and equipment and the new structures were designed to withstand the effects of severe environmental loads. Resulting stresses both in the new and existing structures are within the respective design basis allowable.

NUREG-0612 was determined as not applicable because no safety-related equipment is being added or installed as part of this package. There are no changes to the existing turbine building crane safe load path. The impact on the control of heavy loads is addressed for the installation of the batteries in MR 90-134*E. (SER 92-004)

33. MR 90-134*D (Common), 125 V DC System. The modification replaces panel D01 to increase its fault interrupting capability.

Summary of Safety Evaluation: The description of D01 as given in FSAR Section 8.2.2 is not altered. A temporary panel was installed at column C and 11.8 during the

replacement process to provide power to distribution panels D11 and D12. Sufficient space is maintained to provide access to existing equipment. Installation of the temporary panel is required since the D01 panel functions must remain in service.

The modification provides for the prevention of a failure of D01 from a high electrical fault current through the replacement of the existing D01 panel with a panel that has a higher electrical fault current capability and coordinates with its upstream and downstream equipment. The temporary TD01 panel, the new D01 panel, and the associated cabling are installed to meet the separation requirements of the plant during all phases of installation including final configuration. The panels meet or exceed the requirements for the existing D01 panel. The panels were seismically qualified to function during and after a safe shutdown earthquake.

To avoid an operating reactor trip during the transfer of loads from the existing D01 panel to temporary TD01 panel, the temporary TD01 panel is paralleled with the existing D01 panel. The source of battery charger current is transferred from battery charger D07 to battery charger D09. Thus swing battery charger D09 is used as a normal power supply to the temporary TD01 during the changeout process.

If battery charger D08 is lost on the DC "B" train, during replacement, battery D06 is capable of supplying DC power for a sufficient amount of time to reestablish power from the charger. Likewise, if battery charger D08 and D09 were lost, battery D05 is capable of supplying DC power for a sufficient amount of time to reestablish power from an available charger. The source of charging current to battery D06 or D05 must be restored or the operating unit(s) shall be placed in hot shutdown condition, per TS 15.3.7.A.1.g and TS 15.3.0.

Administrative controls are applied to ensure no work is performed on opposite train safeguards equipment on both units during conditions requiring work on energized DC equipment. Thus, this portion of the modification does not affect any safety analysis.

During live load transfers, power on Unit 2 is reduced to at or below 480 megawatts electric. The possibility of turbine overspeed at or below 480 megawatts electric is greatly reduced. During work in the DC panels, sliders are opened to defeat IOPS. A loss of DC during this work disables the ET trip circuit. Turbine overspeed protection is still provided by the AST trip circuit, the auxiliary governor and the mechanical overspeed trip device. During live load transfers, necessary precautions are taken (i.e., stationing operator at necessary DC panels) to quickly recover DC in the event of a transient. (SER 92-004-02)

34. MR 90-134*E, H, J, and K (Common); RMP 200, IWPs 90-13*E, H, J, and K, 125 V DC System. The design packages install cable termination, modifies control panel 2C20, adds fire detection and security card readers for the new battery and equipment rooms, tests 125 V DC swing battery bus equipment, and installs switchboards D301, D302, D110, D111, and battery D305.

Summary of Safety Evaluation: Since swing battery D305 and its associated D301 bus is required to replace an existing safety-related battery (D05, D06, D105 and D106), the safe shutdown equipment and the rooms in which the equipment is located are secured. Security card readers are installed at the entrances to the safety-related and non-safety-related battery rooms to control access. The fire doors between the battery rooms have door position monitoring indicators installed. The security computer is taken out of service when the revised software is loaded onto the master computer. Testing of the new security card readers is performed.

Automatic, early warning, heat-type fire detectors are installed in the new battery and switchboard rooms.

The design of the swing bus and its associated controls allows the plant to maintain sufficient independence and redundancy and does not create an unreviewed safety question. Cables which can supply swing equipment and at different times can be either safety-related "A" or "B" train were routed in separate raceways to maintain the separation requirements for affected equipment. Cables which were routed in the same fire zone as redundant Appendix R equipment are protected with an approved fire protection envelope system.

The new equipment meets or exceeds the requirements of the existing plant DC system switchboards, batteries and battery chargers. The new equipment was seismically qualified to function during and after a seismic event. Separation is maintained between redundant trains during all phases of the installation including final configuration. (SER 92-004-04)

35. MRs 90-137 (Unit 1), 90-138 (Unit 2), Reactor Coolant System. The modifications replace the pressurizer level transmitters.

Summary of Safety Evaluation: The new transmitters maintain the necessary environmental and seismic qualifications. The transmitters are seismically mounted and the existing impulse line mounting hardware and electrical conduit used. The transmitter replacement occurs when the unit is in a cold shutdown condition.

No setpoint changes, control system functional changes, or changes in the accident analysis parameters are required.

Each system is tested by comparing the output of the level indication with a given calibration input. These comparisons are in addition to performing leak checks and a normal calibration on the system components. The new transmitters provide more accurate and reliable indication. (SER 92-021)

36. MR 90-150*A, *C (Common); RMP 203; and IWP 90-150*A1, *A2, *C, 125 V DC System. The design packages install non-safety-related buses.

Summary of Safety Evaluation: The design packages reduce the loading on the Class 1E batteries by transferring non-1E loads to new non-1E 125 Vdc distribution buses. This is accomplished by installing new non-1E 125 Vdc switchgear, batteries and battery chargers. The existing non-1E loads (emergency turbine lube oil pumps 1P-37D and 2P-37D) are disconnected from D01 and D02, respectively and reconnected to the new non-1E buses 1D201 and 2D201.

MR 90-150*A and *C does not adversely affect the equipment required for safe shutdown of the plant as described in the FSAR.

The worst event that could occur as a result of this modification is a loss of a non-1E DC train and the unavailability of an emergency turbine lube oil pump. The only possible postulated failure, which could impact safety equipment, is if a fault occurs while a non-class 1E bus is connected to the swing bus. However, during this configuration, the swing bus is isolated from the Class 1E buses and systems and the failure does not impact safety-related loads. Since safety systems are not affected by this change, with the exception of the swing battery/bus, this modification does not affect any existing safety analyses. (SER 92-056-01)

37. MR 90-150*B (Common). 125 V AC System. The modification adds HVAC and safety shower/eyewash stations to support the new battery room.

Summary of Safety Evaluation: The HVAC, fire protection and personnel safety systems/components installed via MR 90-150*B are located outside the battery and switchboard cubicles, with the exception of the self-contained eyewash station to be located inside of the Class 1E battery room, and the fire damper/sleeve located in the ceiling slab and walls. The fire damper/sleeve assemblies are seismically installed to prevent interaction with the Class 1E equipment located inside the cubicles; the self-contained eyewash station was relocated to prevent its interaction with Class 1E battery D305 during a seismic event.

The components comprising Class 1E switchboard D301 were examined and determined no catastrophic or sudden failure occurs due to exposure or operation in an elevated ambient temperature of up to 150°F. The manually-operated, molded-case switches contained in the panel are molded-case circuit breakers without trip units (the exceptions are the test and charger breakers). Spurious trip of the test breaker has no safety consequence.

Thermal transient analyses demonstrate it takes approximately 3-3/4 hours for the DC switchboard room, housing D301 to reach 150°F following a loss of all HVAC, assuming the turbine building is at its maximum design condition of 115°F and all equipment in the DC switchboard room is energized and operating at full load. This is sufficient time to take compensatory measures to limit the room temperature rise (e.g., trip battery charger 1D207 and 2D207 to limit room heat load, open doors between the switchboard and battery rooms, provide temporary, portable fans). (SER 92-056)

38. MR 90-160*A (Unit 2). Feedwater System. The design package installs six moisture separator reheater (MSR) dump line drain connections.

Summary of Safety Evaluation: The drain connections provide a method of venting the high-energy fluid from these lines while minimizing the potential for personnel injuries.

The addition of these drain connections does not affect any of the conclusions reached in the FSAR. The drain connections comply with the requirements of B31.1-1967 and other original piping system design requirements. The installation is not seismic and is non-QA. (SER 92-073)

39. MR 90-221 (Common). 125 V DC System. MR 90-221 provides interrupting capability for DC buses D11 and D13 while maintaining selective coordination with downstream breakers and fuses. It installs fuses in series with several thermal-only breakers in D11 and D13, and in cases where complete coordination is not critical, replaces the remaining thermal-only breakers with thermal-magnetic breakers.

Summary of Safety Evaluation: Installation of fuses in series with breakers supplying control power to the safety-related 4160 V and 480 V switchgear, control power is switched between thermal and alternate DC supplies. Control power to both trains is switched so one DC supply does not feed both trains of switchgear. Control power is not available during switching. This is acceptable because of the short duration of the power loss and the ability of auxiliary operations to quickly restore the DC supplies to the normal lineup is necessary.

The loading on a particular station battery is increased slightly during installation, due to supplying normal control power to some switchgear and alternate control power to other switchgear. The increased loading is acceptable due to the relatively small increase in load and its short duration. (SER 91-077)

40. MR 90-227 (Unit 1), 125 V DC System. MR 90-227 replaces the 30 amp tripping circuit fuses for voltage breakers with 15 amp fuses, and installs 15 amp fuses to protect the remote control cables for solenoid-operated breakers.

Summary of Safety Evaluation: Non-conformance report N-89-27 indicated the FSAR Section 7.2.1 design basis requirement was not met. The 4160 V breaker control circuits are fused at 30A and 40A. Control cables for these circuits are size AWG# 12 which is only adequate for 20A circuits. The modification corrects this deficiency for Unit 1 4160 V breakers. An additional 15A fuse for the solenoid closure-type breakers is added to the remote portion.

Some of the 4160 V breakers are safety-related and supply safety-related loads or provide connection to the diesel generators. Reducing fuse size or adding fuses to the control circuits for these breakers could affect the ability for the breaker to open and close to serve a safety-related function. The smaller or additional fuse could blow and keep the breaker from operating. However, the new fuse design is adequate and there is a negligible increased potential for fuse failure to disable breaker operation. Conversely, the new fuse design eliminates FSAR requirement deficiency. If one of the associated control cables was exposed to a high fault greater than its ampacity but less than the fuse rating, it could eventually result in a cable tray fire. Therefore, this revised fuse design improves safety. (SER 91-030)

41. MR 91-010*A (Unit 1), 4160 V Electrical System. The design package replaces the four 10-pole test switches of 1A05 and 1A06.

Summary of Safety Evaluation: The new switches have metal posts to mount the covers requiring no new wiring or lugs. Continuity tests are performed to verify the connections are adequate and the switches are functional. No system functional logic changes were made.

The new switches are installed while safeguards buses 1A05 and 1A06 are out of service during U1R19. The buses are deenergized one at a time for maintenance purposes. While a bus is out of service, its undervoltage protective circuits and emergency diesel supply breaker are not needed.

The modification is QA-Scope due to the impacts on vital switchgear. The new switches are seismically mounted over the existing switch cutouts via a sheet steel mounting plate with machine screws. The seismic adequacy of the switchgear is maintained. (SER 91-079-01)

42. MR 91-032 (Unit 1), Main Control Boards. MR 91-032 replaces the existing rod withdrawal Bank D high (RWBDH) alarm with an automatic rod motion alarm.

Summary of Safety Evaluation: Removal of the RWBDH alarm was previously approved by MR 90-080/081. The RWBDH alarm was provided to address the minimum Bank D bite limit value to ensure adequate negative rate of reactivity addition was available for transients. This is not necessary as operating philosophy is based on operation with rods out since an adequate negative reactivity insertion rate exists at the ARO condition. The setpoint of the RWBDH alarm is based on the bite limit of the Bank D rods and therefore, the alarm is usually continuously lit near the end of core life. Bank D rod position continues to have a high alarm via the PPCS. Operators are notified of automatic rod motion by the clicking of the bank demand centers and the rod motion indication lamps. The new alarm provides notification to the operator via a standard main control board alarm. This alerts the operator immediately to unplanned reactivity changes.

The alarm is generated by the use of spare contacts in the bank selector switch and the rod motion indicating lamp circuit. The bank selector switch contacts provide indications that automatic mode was selected and the rod motion indication lamps are lit during any rod motion. By arranging these signals to drive a relay, a contact closure required for the alarm can be generated. The 100 V DC relay used to generate the contact closure is mounted in the position left vacant by the removal of the RWBDH relay. The function of the RWBDH alarm is also fulfilled by this new alarm since the RWBDH alarm is intended to prevent rod motion withdrawal beyond the all rods out position while in the automatic rod control mode. The new alarm is installed during unit shutdown when the rod control system is deenergized. The alarm is tested following installation by simulating an automatic rod motion signal.

The new alarm improves the operators awareness of an automatic rod motion. Faults in the addition to the rod motion relay circuit produces the same effects as faults in the rod motion indicating lamp circuits. These faults produce alarms (rod control system non-urgent and urgent failure) which clear as soon as automatic rod motion ceases. (SER 91-081)

43. MR 91-037*A (Common), Condensate System. MR 91-037*A installs a dike across the north end of the Unit 2 turbine hall condenser pit. The dike is located south of the northern end of the condenser pit and runs east/west across the entire width of the pit. The dike will contain sodium bisulfite in the event of a rupture or leak from the sodium bisulfite tank located in the condenser pit.

Summary of Safety Evaluation: The dike does not have an affect on equipment related to safe operation of the plant or safe shutdown equipment. A sodium bisulfite spill produces non-toxic fumes that are skin, eye and respiratory irritants. The control room habitability during toxic gas conditions was evaluated and determined that the control room environment was safe upon providing additional self-contained breathing apparatus (SCBA) units for control room personnel. The number of SCBAs in the control room has increased.

It was more desirable to pump the sodium bisulfite to the retention pond and then to Lake Michigan, than to hold it in the condenser pit while pumping it to barrels or a truck or neutralizing it. This minimizes personnel exposure to the sulfur dioxide fumes given off by the sodium bisulfate.

The wet fire suppression system is located in the Unit 2 turbine hall condenser pit. The greatest potential for a fire on El. 8' is an oil fire subsequent to an oil spill from the oil storage tank located on the south side of the turbine hall. The purpose of the system is to suppress fire in its initial stages. The retaining dike installation does not violate fire system protection area requirements that are stated in NFPA Handbook Chapter 13. The dike sections off the northernmost portion of the condenser pit, protecting the components of the dechlorination system from potential spills of liquids in other areas of the condenser pit. The dike does not interface with fire protection equipment described in FSAR Section 9.6. There is no adverse impact on the fire suppression system located in the Unit 2 condenser pit.

During the construction of the retaining dike, flame retardant lumber is used. Therefore, the possibility of a fire is not increased. (SER 91-107)

44. MR 91-065 (Common), Main Control Boards. MR 91-065 installs two switch guards onto the rear of main control board C02 to prevent inadvertent operation of the switches for the 480 V bus feeder breakers and gas turbine breakers.

Summary of Safety Evaluation: Both guards are seismically fastened to the control panel to prevent damage to any adjacent equipment during a seismic event. Because these guards are designed to not interfere with the intended operation of the switches or the control board, the modification has no effect on the safety and operation of the plant. This guard however, decreases the possibility for accidental switch operation if bumped. (SER 91-112)

45. MR 91-067 (Common), 125 V DC System. The modification increases battery chargers D07, D08, and D09 ground alarm circuitry sensitivity.

Summary of Safety Evaluation: The existing resistor/relay network on D07, D08, and D09 detects the DC system ground, and a contact on the relay activates the appropriate battery charger alarm on ASIP 2C20. This network is replaced with an individual circuit card which is connected across the positive and negative DC leads for each of the affected chargers. A contact on the circuit card relay feeds into an auxiliary relay, which is also fed by the undervoltage relay on the chargers. The auxiliary relay alarms on a ground detection alarm, or on loss of power to the ground detection circuit as indicated by the undervoltage relay. This auxiliary relay feeds into the same annunciator circuit and activates the same alarm on ASIP 2C20 for each charger.

The ground detection circuit is fused on both the positive and negative legs with 1 amp panel-mounted fuses. The fuses isolate the detection circuit from the main charger output and provide an easy disconnect to the circuit during the performance of troubleshooting/DC ground location activities with portable ground location equipment. The small size of the fuses does not pose a seismic concern, as they are not likely to be dislodged from their mounting location in the door of the charger panel and fall into the cabinet on top of the enclosed equipment during a seismic event.

The setpoint for each card supplied was shop verified and certified to be a minimum of 16,000 ohms. Documentation showing the seismic capability of the circuit card design was provided. The seismic adequacy of the mounting design for the PCP card was evaluated and found to be acceptable per Seismic Class 1 criteria as documented in calculation N-91-087.

The modified ground detection circuits on D07, D08, and D09 are more accurate and reliable as evidenced by the reliable operation of the identical ground detection circuit in use on D107, D108, and D109. This increases the reliability of the DC system and the equipment it supplies.

Installation of this modification is procedurally controlled. The ground detection circuit in swing battery charger D09 is replaced, and chargers D07 and D08 are removed from service one at a time while charger D09 is brought on-line to replace each battery charger while out of service. Swing battery charger D109 is brought on-line to replace charger D108 while it is out-of-service. (SER 92-005)

46. MR 91-086 (Unit 2), Computer System. MR 91-086 installs necessary wiring to make intermediate range startup rates available to the computer. This involves disabling the rate and comparator drawer in 2C133 during installation, and results in a loss of source and intermediate range startup rate indication. It also causes loss of power range comparators and generates a power channel deviation alarm.

Summary of Safety Evaluation: MR 91-086 corrects a deficiency discovered in the safety assessment system critical safety function subcriticality status tree. This modification allows continued use of the SAS/SPDS system by providing the necessary intermediate range startup rates to the computer multiplexers. This involves tapping off

existing signals in the rate and comparator drawer of NIS cabinet 2C133 and terminating existing wiring from the computer multiplexers.

This modification provides no new hardware functionality. The safety concerns are during the period of installation. The rate and comparator drawer are disabled. This drawer contains the source and intermediate range startup rate circuitry, and the power range comparator circuitry. The result of disabling this drawer is loss of all startup rate indication, and generation of a power channel deviation alarm. N31R, N32R, N35R and N36R, however, startup rate indication is still available from N40.

Performing the modification during U2R18 when there is no fuel movement, ensures the unit is in a state where these indications are not required and the deviation alarm can be ignored. (SER 92-097)

47. MR 91-108*A (Common), Fuel Oil System. MR 91-108*A installs a drain/sample line in each fuel oil tank drain line isolation valve bank flange. This provides separate samples for each tank, and a means to drain water from the bottom of the tanks.

Summary of Safety Evaluation: The drain/sample line provides local sample points for each fuel oil storage tank (FOST). Samples are taken after delivery, and during the months of June, and September through March. The line also provides for the lowest local draining point for the FOSTs. The drain/sample line is installed on the drain line isolation valve existing blank flange. The components of the drain/sample line were selected to meet the pressure and temperature ratings of HB-22, which is the piping specification for the fuel oil system. The materials are stainless steel for corrosion resistance since no protection from outside elements is provided. The FOSTs are not taken out of service during the installation since the drain line can be isolated by the drain line isolation valve.

The drain/sample line does not adversely affect FOST operation if a failure occurs. A failure associated with the drain/sample line would be a leak, caused by water freezing in the lines that might be present. This would leak fuel oil from the FOSTs to the drainage dyke, which would contain the fuel oil in one area. The level of the tanks is logged each shift, and has a low level setpoint alarm (>78%). (SER 91-090)

48. MR 91-133*A (Unit 2), Boric Acid Transfer System. MR 91-133*A installs monitoring instrumentation necessary to trend boric acid transfer pump performance as required by the inservice test program (IST). It includes a magnetic flow tube and transmitter, a suction side pressure gauge, modification of an existing boric acid filter inlet pressure gauge to allow calibration for use as a pump discharge pressure gauge, and a removable pump enclosure to allow manual vibration monitoring.

Summary of Safety Evaluation: The installation is performed in accordance with original design specifications. The components have no affect on the operation of the boric acid system. The low weight of the additional components does not affect the seismic capability of the system. Post-modification testing documents the new IST instrumentation does not impact system operation. The flow tube is installed just upstream of the tee separating the normal and emergency flow paths. This allows the flow transmitter to be used as a local indication of flow whenever desired by the control room. Installation occurs during a Unit 2 refueling shutdown. The interim conditions during the shutdown isolates the system to allow it to be flushed and drained and heat tracing deenergized. In this condition, it is not possible to add concentrated boric acid using the normal boration path. A boration flow path from a boric acid storage tank or refueling water storage tank to the reactor coolant system through the safety injection pumps exists at in accordance with Technical Specification requirements. The work scope was reviewed as a part of the shutdown risk assessment. (SER 92-093)

49. MR 91-149*B (Common), Service Water System. MR 91-149*B corrects deficiencies in the service water piping support structures. These supports are currently documented as operable in support report calculations 300023, 300024, 300025, 300026 and 300027 but did not meet the applicable Code allowables. The support corrections do not interrupt the functioning of the service water system.

Summary of Safety Evaluation: Piping supports identified as integral welded attachment (IWA) supports are supports that require welding to the piping pressure boundary. The welding of these supports shall not compromise the pressure boundary and do not impact the operation of the plant. After removal of the existing IWA attachment from the piping, the wall thickness shall be checked to assure adequate wall thicknesses for pressure retaining purposes. Additionally, the metal temperature shall be verified to assure exceedance of the minimum temperature required by the 50°F per ASME Section 9 and AWS D1.1, or pre-heating of the piping shall be performed. Welding to the exterior of the pressure boundary in this manner is consistent with standard industry practice.

Temporary piping supports of an industrial grade nature (chain falls and stands) are used to ensure the piping system affected by the specific support work survives a DBE within the JCO code allowables. Thus, the system is considered operable throughout the modification. The amount of support capacity required is identified in the installation work plan. Temporary support configuration(s) are reviewed by the responsible engineer or installation supervisor for adequacy. (SER 91-108)

50. MR 91-156 (Unit 1), 480 V Electrical System. MR 91-156 corrects the train separation conflicts involving the control circuits of 480 volt non-safeguard load breakers 1B52-12B (safeguards bus 1B03), 1B52-21B, 1B52-22A and 1B52-22C (safeguard bus 1B04).

Summary of Safety Evaluation: These circuits were reclassified as safety-related because a failure in the circuits could prevent their respective breakers from tripping under safety injection (SI) or safeguard bus undervoltage conditions as required by the FSAR. One of these raceways could take out one of the trains and also the control power fuses of one of these loads on the opposite train preventing that load from being stripped. The cable conflicts are resolved by the installation of branch fuses on the portions of these circuits remote from their respective bus section. The branch fusing is coordinated with the breaker main 10 amp control fuse so a failure of the remote portion does not take out the 10 amp fuses. This ensures the breakers retain control power to allow the units to trip as required. Failure of the branch fuses is indicated by loss of status lights or 86 relay seal-in function of in pressurizer heater controls.

Since the SI trip signal presently is brought from the control room via the same remote circuits, an auxiliary relay is installed for each train, in 1B03 and 1B04, to supply a secure SI trip signal to these breakers. These relays are connected to the SI lockouts in the control room via existing secure control cables to a future breaker position in each bus section. The auxiliary relays and their power supplies are monitored by a coil status on the cubicle door. The status light is added to the turbine building shift log so it is checked every shift.

The breakers serve non-safeguards loads. Specifically, the loads involved are the pressurizer heaters, which are not used during cold shutdown, and the service air compressor, which has no safety significance. As such, removing them from service for the modifications and testing during cold shutdown presents no safety concerns. (SER 92-029)

51. MR 91-157 (Unit 2), 480 V Electrical System. MR 91-157 corrects the train separation conflicts involving the control circuits of 480 volt non-safeguard load breakers 2B52-36B (safeguards bus 2B03), 2B52-29B, and 2B52-30C (safeguards bus 2B04).

Summary of Safety Evaluation: The separation conflict arose when these circuits were reclassified as safety-related because failure in these circuits could prevent their respective breakers from tripping under safety injection or safeguard bus undervoltage conditions as required by the FSAR. The cable conflicts are resolved by installing branch fuses on the portions of these circuits remote from their respective bus section. The branch fusing is coordinated with the breaker main 10 amp control fuse so a failure of the remote portion does not take out the 10 amp fuses. This ensures that the breakers retain control power to allow the units to trip as required. Failure of these branch fuses is indicated by loss of status lights or 86 relay seal-in function of pressurizer heater controls.

The breakers serve non-safeguard loads. Specifically, the loads involved are the pressurizer heaters, which are not used during cold shutdown. MR 91-157 has no negative impact other than a higher probability of an outage to these non-safeguard loads due to nuisance fuse failures because of the additional fuses in these circuits. This higher probability is not considered to be significant. (SER 92-080)

52. MR 91-162 (Unit 1), Engineered Safeguards Feature (ESF) System. MR 91-162 upgrades miscellaneous relay rack 1C158 from non-safety-related to a safety-related and QA classification. This modification enhances the structural integrity of 1C158 and engineered safeguards consoles 1C167 and 1C166. These objectives are accomplished by installing a new intercabinet structural brace between cabinet 1C158 and 1C166, and a new base anchorage of cabinet 1C158 and adjacent cabinets 1C167 and 1C166.

Summary of Safety Evaluation: The cabinets are modified as a result of non-conformance report N-90-217. 1C158, 1C167, and 1C166 shall not be taken out of service or deenergized to perform the work outlined in MR 91-162. Although some of the equipment within these cabinets is important to safety, the work is performed during a refueling outage. TS 15.3.5 allows engineered safeguards to be inoperable at this time. ICP 10.1, which is performed prior to refueling shutdown, bypasses and blocks safeguards and AMSAC systems to prevent accidental actuation of these systems. ICP 10.1 is in effect prior to the start of this modification and during installation of the modification. Startup testing, which is normally performed prior to reactor startup to verify proper operation of the safeguards and AMSAC systems, is performed after completion of the modification. Furthermore, precautions are taken to prevent any metal filings or other debris from falling on, or into equipment located within the affected cabinets. Therefore, operability of 1C158 and engineered safeguards consoles 1C167 and 1C166 and equipment important to safety in the location of the upgrade is unaffected. (SER 91-073-01)

53. MR 91-202 (Common), Service Water System. MR 91-202 installs a 3/4" pressure tap on the 20" HB-19 pipe. The installation is approximately 2' downstream of butterfly valve SW-165.

Summary of Safety Evaluation: The line is isolated during installation to minimize service water leakage through the new tap; however, the line cannot be fully drained so minor leakage is expected. Isolation of this line includes the "A" component cooling (CC) heat exchanger (HX-12A). The "A" CC heat exchanger, though isolated, is not considered out of service, but is considered available for service if needed. Opening of valves SW-165 and SW-286 is performed as soon as possible after the installation to return it from isolation. During installation, the "D" CC heat exchanger is out of service as a result of conditions not related to this modification. However, HX-12B and HX-12C

are both in operation and receiving service water supply from the north service water header and are dedicated to Unit 1 and 2 component cooling respectively, and HX-12A is on standby, (swap HX-12A and HX-12B with HX-12A dedicated to Unit 1 and HX-12B available for either unit) and available for component cooling; therefore, three heat exchangers are available and no LCO is entered. The "A" battery room cooler (HX-105A) normally supplies service water from the south service water header; however, as a result of the isolation of the 20" service water pipe, it is supplied from service water north header. Technical Specification limits are not exceeded for the component cooling or the service water systems. (SER 91-115-01)

54. MRs 91-204 (Unit 1), 91-205 (Unit 2), Chemical and Volume Control System. The normal charging line flow control valve for each unit, CV-142, was added to the inservice test program. The program requires the annual fail safe test and stroke test open of CV-142. The modification adds a Whitey 3-way ball valve in the operating air line between the valve positioner and the diaphragm operator.

Summary of Safety Evaluation: The 3-way ball valve and its three positions (normal, shut, vent) are labeled. It can be locked in one position - "normal." During normal operation of CV-142, the 3-way ball valve is locked in the "normal" position and the vent connection capped.

The modification slightly increases the mass of CV-142, (a Seismic Class I AOV). The resulting additional forces transmitted to and the effect upon the AOV and its supports during a seismic event is negligible.

The installation does not increase the probability of the valve positioner to valve operator air line failure. Should the air line fail, the consequences are no different than those originally designed for; the valve fails open.

Such a failure is evident to the control room operator by indications on the control boards. While repairs are made existing instrument air valves can be used to isolate the air line, CV-323B can be used to regulate normal charging line flow, and the charging pump speed can be adjusted to maintain seal injection flow to the RCP seals.

Installation of the modification does not change the normal operation of CV-142, while it allows the required testing to be performed during cold shutdown. Its failure causes the air-operated valve (AOV) to go to its original design for fail safe position, which is evident at the control boards, and can be dealt with by use of the AOV bypass, CV-323B, and by controlling charging pump speed. There are no adverse system interactions. (SER 92-018)

55. MR 91-210*A (Unit 2), Main Steam System. MR 91-210*A modifies the main steam isolation valves by installing new valve components, which both reduces the amount of force required to fully shut the valves, and increases the forces available to fully shut the valve.

Summary of Safety Evaluation: The 12" cylinder on 2MS-2017 is replaced with a new 12" cylinder with a stronger spring which reduces the amount of opening force available in this valve. Only the valve components are affected. The control and function of the valves are not impacted. The design was performed by the valve manufacturer, and accounts for potential failure mechanisms. The valve manufacturer has successfully performed this modification at two other nuclear plants, and the new configuration has proven to be effective. Reviews performed confirm the design incorporates items that improve valve performance, and do not increase the potential for valve failure. The new design does not impact the ability of the valve to shut tightly under steam pressure as required by the steam generator tube rupture accident analysis. It improves the ability

of the valve to shut within 5 seconds as required by the steam pipe rupture accident analysis and technical specifications.

MR 91-210*A is performed during cold shutdown conditions. In addition, the steam generator secondary side manways and handholes are installed when containment integrity is required. (SER 92-079)

56. MRs 91-219 (Unit 1) 91-220 (Unit 2), Auxiliary Feedwater System. MRs 91-219 and 91-220 cut and cap the turbine-driven auxiliary feedwater pumps (TDAFP) P29 governor sensing lines. The sensing line is isolated from the AFP room, administratively keeping valves 2MS-245 and 2MS-132A shut. The regulator is disabled by moving the travel stop so it cannot contact the governor linkage lever.

Summary of Safety Evaluation: The removal of the governor sensing lines from the turbine controls has no effect on pump operability as proven in the engineering evaluation for MRs 91-219/220. In the engineering evaluation, several fast start tests were performed on 1&2P29 with the governor sensing line disabled and RPM traces made. These RPM traces were compared with those taken with the governor sensing line enabled. It was concluded that the governor sensing line does not impact the ability of the governor itself to prevent overspeed. The turbine manufacturer was contacted and stated that governor sensing lines not needed may be disconnected without other required modifications.

Socket weld pipe caps are installed in accordance with B31.1-1967 and original construction material specifications. Removal of the line and addition of the pipe caps at the root connections just downstream of root valves 1&2MS-251A and 252A does not degrade the seismic integrity of the steam supply piping to 1&2P29 or of the root valves. After the governor sensing line is removed, only the steam trap piping installed by MR 89-034 remains downstream of these root valves. A review of MR 89-034 showed no credit was taken for the governor sensing line supports in evaluating the seismic integrity of the root valves.

The line removal work is performed when the affected unit is in a cold or refueling shutdown when the TDAFP is not required. Controls are specified in the installation work plan so the work does not jeopardize the operability of equipment important to safety (i.e., fire barrier penetration, safe shutdown area scaffolding, transient combustible and cleanliness procedures). The pipe caps are pressure tested in accordance with the pressure test program. The monthly inservice tests of 1&2P29 are performed as a final acceptance test prior to reactor criticality to ensure that the work around and to the governor valve linkage lever has no affect on pump operability. Since the engineering evaluation has already demonstrated that removal of the governor sensing line does not affect pump operability, further fast start testing is not required for modification acceptance. (SER 92-019)

57. MRs 91-224 (Unit 1), 91-225 (Unit 2), Main Steam System. MRs 91-224/225 install valve position indication for non-return valves 1&2MS-2017A and 1&2MS-2018A.

Summary of Safety Evaluation: Valve position indication provides a positive and accurate indication when the valve is fully shut and an approximate valve position for when the non-return valves are off their closed seat. This makes the valve position determination process much easier during valve testing or shutdown verification requirements. The valve position indicator is installed flush with the gland packing flange and allows for the use of live-load packing and packing adjustment. The rotation and operation of the valves does not affect the valve position indicator. Valve position is determined by aligning the pointer attached to the non-return valve counterweight arm with the markings on the scale. (SER 92-042)

58. MRs 91-226 (Unit 1) 91-227 (Unit 2), Nuclear Instrumentation System. MRs 91-226/227 eliminate a turbine runback and rod withdrawal block as protective actions in the event of a dropped RCCA signal generated by the rod bottom bistables or by the negative flux rate circuits of the nuclear instrumentation system. Eliminating these protective actions decreases the number of inadvertent actuations of the turbine runback.

Summary of Safety Evaluation: Eliminating the turbine runback and rod withdrawal block as protective actions in the event of a dropped RCCA signal does not involve an unreviewed safety question. The Technical Specification Basis on instrumentation systems is changed to reflect the new configuration. Limiting conditions and testing requirements in the Technical Specifications remain unchanged.

Analyses performed by the NSSS vendor showed that the protective actions being eliminated are not required to prevent departure from nucleate boiling (DNB) in the core. Plant response to a dropped RCCA changes as a result of the modification. RCS pressure and temperature decrease due to secondary side demand in excess of primary side power after the RCCA drops. Reactor trip due to low pressurizer pressure, or turbine runback due to OTΔT could potentially occur. The reactor trip is designed to protect the core from excessive voiding. The OTΔT turbine runback is designed to maintain margin to the OTΔT reactor trip. Maintaining margin to the trip gives the operators time to adjust the flux shape before a trip occurs. The occurrence of a reactor trip, turbine runback or new steady-state condition of the plant was evaluated and does not increase the consequences or reduce the margin of safety of a dropped RCCA event.

The modification is performed while in a shutdown condition and post-modification testing is required to minimize the potential for equipment malfunction. (SER 92-017)

59. MR 91-229 (Unit 1), Feedwater System. The modification replaces the nozzles for the extraction steam inlets to the shell side of #4A and #4B low pressure feedwater heaters HX-20A&B.

Summary of Safety Evaluation: The modification is installed during the steam-out phase of refueling shutdown. The feedwater heaters are out of service during the duration of the installation. Materials, procedures and installation activities are in accordance with Section VIII. The Wisconsin Administrative Code requires State permission be granted in accordance with Form S-90. This form and the alternation plate for the vessel is obtained by the installation group. The authorized inspector was notified of the work and ensures all State and Code requirements are met. After installation, a hydrostatic pressure test of the shell side of the heat exchangers is performed to ensure original pressure ratings are maintained.

The new nozzles are the same dimensions as the original design. Original requirements for pressure rating are maintained. Because of the resistance of Type 304 stainless steel to erosion, reliability increases. (SER 92-016)

60. MR 91-235 (Common), Waste Gas System. MR 91-235 removes the rupture disc from T20A waste gas decay tank. A short length of pipe replaces the disc and associated flanges.

Summary of Safety Evaluation: The rupture disc was added in line with pressure relief valve WG-1621 to allow this tank to be used in the cryogenic absorber package. The rupture disc function in this application is to seal the tank, while the relief valve is for protection of the tank. The cryogenic absorber package was not licensed and therefore, tank T20A is not needed for use in this package. This tank was returned to normal service under MR M-708. During a recent 10-year hydrostatic test performed on

T20A&B, the rupture disc was removed from the rupture disc assembly. There were no replacement rupture discs available at the time of the test. Therefore, since the rupture disc is no longer needed to support the cryogenic absorber package, it was removed. MR 91-235 was designed, installed, and tested to the criteria of B31.1-1967.

The installation takes place while T20A is tagged out of service. The tank remains in this configuration throughout installation. (SER 92-009)

61. MR 92-009 (Unit 1), Main Steam System. MR 92-009 changes the operating mechanism of the main steam isolation valves (MSIVs) to reduce the friction opposing the closing force of these valves. This is done to minimize the potential for the valves sticking open.

Summary of Safety Evaluation: Packing friction has proven to be a contributing factor to past valve failures to shut. One change is to replace the outer packing rings with a new composite graphite material and to reduce the total number of rings used on the valve shaft. The composite material provides 25-50% less friction than braided graphite rings. This type of packing was used successfully at other nuclear plants in other valve applications. The changes reduce the total packing friction and make the valves less susceptible to sticking.

Initial packing loads are set in a cold condition. This is done by increasing packing gland torque in 10 ft-lb increments, and stroking the valve after each increment and is continued until the valve begins to stick open. If excessive steam leaks are discovered during startup, the above iteration is repeated with the plant in the hot standby condition. (At higher temperatures, packing generally exhibits less friction. Additionally, higher pressures in the steam generators will assist valve closure.) This allows higher packing loads in the hot condition, in order to more positively prevent steam leaks through the packing.

One precautionary measure is to install a shield on the air cylinder and/or linkage, to keep moisture out of the cylinder, should a steam leak eventually redevelop. The shields are made of light gauge sheet metal and have been field fit so they do not interfere with valve operation. The shields are reinstalled on the valves after the valves are reassembled during U1R19 outage.

Tests are performed after their installation, in order to ensure they do not interfere with valve stroke times (TS requirement of 5 seconds or less). (SER 92-025)

62. MR 92-009 (Unit 1), MWR 923025, MWR 923043, Main Steam System. The modification increases clearance between the valve shaft and packing gland follower.

Summary of Safety Evaluation: MR 92-009 removes sufficient material from the inside of the main steam isolation valve (MSIV) packing gland follower so it becomes physically impossible for the packing gland follower to come in contact with the MSIV shaft when installed. Inspection evidence following disassembly of 1MS-2017 and 1MS-2018 at the end of U1R19 showed that MSIV shaft galling had occurred due to contact between the MSIV shaft and packing gland follower. This galling was the primary cause of the failure of 1MS-2018 during stroke testing at the end of U1R19. Modification to the packing gland followers prohibits MSIV shaft galling, therefore, reducing the probability of valve malfunction.

The function of the MSIV packing gland follower is to transmit pressure to the MSIV shaft packing. The modification does not affect this function, or the ability of the packing gland follower to perform this function in any manner whatsoever. Additionally,

the function of the MSIV and its ability to perform that function is, likewise, not affected in any manner. (SER 92-025-03)

63. MR 92-009-01 (Unit 1), Main Steam System. MR 92-009-01 increases the size of the vent hole on the upper portion of the main steam isolation valve (MSIV) operators from 1/8" to 1/4". This allows the use of a fiberscope for internal inspection during operation. Scoping allows for internal inspection to determine where and when the water is appearing in the operator.

Summary of Safety Evaluation: A small 1/8" hole exists in the top half of the MSIV operating cylinder. In order to inspect the internals with a fiberscope, the hole is enlarged to 1/4" diameter. The hole functions as a bleed port when the operating piston rises, and as a vacuum breaker when the piston descends. The increase does not structurally affect the operating cylinder. The enlargement allows the valve to operate more efficiently as the differential pressure across the hole decreases. Enlarging the vent increases the diagnostic capability to ensure proper operation of the MSIV cylinders.

This change does not significantly increase the probability of getting foreign material or moisture in the cylinder. It does not significantly change the dampening affect when the cylinder piston strokes.

This change is only to be performed on 1MS-2018. 1MS-2017 was reinstalled; therefore, a risk of getting metal shavings inside the cylinder exists. (SER 92-025-01)

Summary of Safety Evaluation: This revision changes the scope of the MSIV work to include both MSIVs (SER 92-025-01 stated only 1MS-2018 would be worked). The operators are disassembled prior to drilling the new hole in order to avoid getting metal shavings inside. (SER 92-025-02)

64. MR 92-010 (Common), Fire Protection System. MR 92-010 modifies the computer room halon panel to allow surveillance testing of the pre-alarm circuit.

Summary of Safety Evaluation: The computer room halon fire detection system is provided for property protection purposes only. The manner in which the control room receives and acknowledges alarms changes. The modification does not impede control room alarms from any safe shutdown fire detection panel.

There are no Appendix R-related design concerns associated with the modification. (SER 92-031)

65. MR 92-017 (Common), Gas Turbine. The modification replaces the existing totalizers with a new pneumatic control system.

Summary of Safety Evaluation: The concerns with the gas turbine overhaul and upgrade are: 1) Meeting Appendix R and station blackout (SBO) requirements while G05 is out of service; and 2) Incorporating the manufacturer recommended upgrades for Model W251A gas turbines to achieve 95% reliability using proper work practices so equipment important to safety is not impacted.

The first concern was addressed by the temporary diesel generator installation. While G05 is out of service, the temporary diesel generator (TDG) is used as a power source for SBO and Appendix R alternate shutdown scenarios. Installation and siting of the TDG was addressed in SER 92-049.

During the overhaul, G05 is disconnected from tie bus H01 and tagged out along with most of its supporting systems in accordance with the equipment isolation procedure to allow work on these systems. The gas turbine building sprinkler system was tagged out. This action was reviewed and approved by the Fire Protection and Safety Coordinator. The gas turbine building is isolated from the rest of the plant with the exception of its close proximity to the X04 transformers. The overhaul-related activity which could impact the X04 transformers is physical contact with the crane which is brought in to assist in the overhaul. Crane controls are in accordance with internal procedures. The loss of the X04 transformers was previously analyzed.

Post-overhaul testing consists of post-modification testing as applicable, a pre-startup control and electrical checkout, and monitoring of critical parameters upon startup prior to loading the machine. Operational testing is performed in accordance with established procedures.

The overhauled gas turbine is an enhancement of the existing unit and is not fundamentally different. (SER 92-062)

66. MR 92-025*A (Unit 1), IWP 92-025*A, Main Control Boards. MR 92-025*A provides a means to perform meter calibrations on: 1X13/X14 ammeters, 1A05/A06 voltmeters, 1B03/B04 voltmeters, 1Y01/Y02/Y03/Y04 voltmeters, and 1Y101/Y102/Y103/Y104 voltmeters.

Summary of Safety Evaluation: The 1A03 voltmeter is connected to the same terminal block as 1A05 voltmeter, and the 1A04 voltmeter is connected to the same terminal block as 1A06 voltmeter, so these circuits are affected in the same way as the 1A05 and 1A06 voltmeter circuits. Several other 4160 V bus voltmeter circuits share a common ground with these voltmeters, so temporary ground connections are provided for these circuits during installation activities.

The 1B03 and 1B04 voltmeters are connected to solid-link terminal blocks. These circuits are rewired to existing sliding-link terminal blocks. The sliding-link terminal blocks are not likely to fail, and if they do, the loss of bus voltage indication does not affect the bus itself. Installation activities for these voltmeters are performed during the 1A05 and 1A06 bus outages. During the 1A05 outage, 1B03 is cross-tied to 1B04, and during the 1A06 outage 1B04 is cross-tied to 1B03. Even though the 1B03 bus is powered while its voltmeter is out of service, the 1B04 voltmeter is in service and provides indication for both 1B03 and 1B04 since they are essentially the same bus. The same is true for 1B04 while 1A06 is out of service.

The following meters are already connected to sliding-link terminal blocks, so no change to the circuits is necessary: 1Y01/Y02/Y03/Y04 voltmeters and 1Y101/Y102/Y103/Y104 voltmeters. These meters are taken out of service one at a time to perform meter face replacement and meter calibration. The loss of bus voltage indication does not affect the bus itself. Further, the operators still have alarm indication of instrument bus over/undervoltage and inverter trouble, so the lack of bus voltage indication presents minimal impacts.

Meter face replacement and meter calibration are also performed on 1B01 and 1B02 voltmeters. These are also taken out of service one at a time to perform this work. (SER 92-036)

67. MR 92-025*B (Unit 2), IWP 92-025*B, Main Control Boards. MR 92-025*B provides a means to perform meter calibrations on: 2X13/X14 ammeters, 2A05/A06 voltmeters, and 2B03/B04 voltmeters.

Summary of Safety Evaluation: The test switch operates as a make-before-break device, so the current path is redirected back to the current transformer before the circuit to the ammeter is opened. The test signal is then applied at the internal wire side of the test switch. These test switches were evaluated and were found to be acceptable for use. The inherently rugged design of the test switch means a failure is unlikely, and a failure is essentially not any more likely than failure of the solid-link terminal blocks to which the ammeter circuits are presently connected.

Installation work for the 2X13 and 2X14 ammeter circuits is performed while the current transformers, which supply these circuits, are deenergized. This is accomplished by performing 2X13/X14 ammeter switch installation in conjunction with routine maintenance procedure (RMP) 23B.

The 2A05 and 2A06 voltmeters are connected to solid-link terminal blocks, which are replaced with sliding-link terminal blocks. The sliding-link terminal block installed is a lot-numbered item, and is identical to the sliding-link terminal blocks used elsewhere in main control board C02 in similar applications. The new terminal blocks are mounted in an identical fashion to these other sliding-link terminal blocks so the seismic qualification is maintained. The sliding-link terminal blocks are rugged in construction, and are no more likely to fail than the solid-link terminal blocks. (SER 92-036-01)

68. MR 92-030 (Unit 1), Auxiliary Feedwater System. MR 92-030 upgrades the turbine-driven auxiliary feedwater pump (TDAFP) steam exhaust piping supports to meet seismic analysis requirements. The requirement to upgrade this system was a result of seismic analyses performed to resolve non-conformance report N-90-210 and the auxiliary feedwater vertical slice audit. The modification includes adding support components to the system.

Summary of Safety Evaluation: The as-built piping and supports for the TDAFP steam exhaust was evaluated for their ability to withstand the loads and stresses imposed on the system by a seismic event. The results of the analysis show that, while the system is operable as is, some relatively simple piping support upgrades increase the system capacity to withstand these loads and stresses. The supports identified are upgraded and a new restraint (at existing support HB15-H5) added to the system. The current as-built operability and the proposed modification acceptability are documented in stress reports 100073 and 100073s, respectively.

The piping support upgrades are: HB15-H1, stanchion support modified to prevent uplift during a seismic event; HB15-H2, spring support adjusted to eliminate loads from the (EB-8) steam supply piping; HB15-H3, wall penetration support modified to meet support allowables; HB15-H4, U-bolt guide modified into a box guide to meet support allowables; and HB15-H5, spring support to add lateral restraint (box guide) to eliminate overload on an inaccessible U-bolt upstream.

The modification is performed during the U1R19 outage with 1P29 out of service, therefore, no LCO is entered. Temporary restraints are provided for HB15-H2 and HB15-H4 during modification and are controlled by the installation work plan. When complete, the piping system meets all applicable Code requirements for seismic piping and supports. (SER 92-020)

69. MR 92-031 (Unit 2), Auxiliary Feedwater System. MR 92-031 upgrades the 2P29 turbine-driven auxiliary feedwater pump (TDAFP) steam exhaust piping supports to meet seismic analysis requirements.

Summary of Safety Evaluation: The as-built piping and piping supports for the 2P29 TDAFP steam exhaust were evaluated for their ability to withstand the loads and

stresses imposed on the system by a seismic event. The results of the analysis show that, while the system is operable as is, some relatively simple piping support upgrades increase the system capacity to withstand these loads and stresses. The supports involved are upgraded and an additional restraint (at existing support HB15-2H5) added to the system. The as-built piping and piping supports and the proposed modification are documented in stress reports 200063 and 200063s, respectively.

The piping support upgrades are: HB15-2H3, wall penetration support modified to meet support allowables; HB15-2H4, U-bolt guide modified into a box guide to meet support allowables; and HB15-2H5, spring type support to add lateral restraint (box guide) to eliminate an overload at an inaccessible piping support upstream.

The modification is performed during the U2R18 refueling outage when 2P29 is out of service, therefore, no LCO is entered. Temporary restraint is provided for HB15-2H4 during modification and controlled by the installation work plan. When complete, the piping system meets all applicable Code requirements for piping and supports. (SER 92-069)

70. MR 92-081 (Common), Gas Turbine. MR 92-081 modifies the main fuel pump recirculation piping which prevents tripping of the pump because of low suction pressure.

Summary of Safety Evaluation: While G05 is out of service, a temporary diesel generator (TDG) is used as a power source for SBO and Appendix R alternate shutdown scenarios. Installation and siting of the TDG was addressed in SER 92-049.

During the overhaul, G05 is disconnected from tie bus H01 and tagged out along with most of its supporting systems in accordance with the equipment isolation procedure to allow work on these systems. The gas turbine building sprinkler system was tagged out. This action was reviewed and approved by the Fire Protection and Safety Coordinator. The gas turbine building is isolated from the rest of the plant with the exception of its close proximity to the X04 transformers. The overhaul-related activity which could impact the X04 transformers is physical contact with the crane which assists in the overhaul. Crane controls are in accordance with internal procedures. The loss of the X04 transformers was previously analyzed.

Post-overhaul testing consists of post-modification testing as applicable, a pre-startup controls and electrical checkout, and monitoring of critical parameters upon startup prior to loading the machine. Operational testing is performed in accordance with established procedures.

The overhauled gas turbine is an enhancement of the existing unit and is not fundamentally different. (SER 92-062)

71. MRs 92-087 and 92-087-01 (Unit 1), Safety Injection System. MR 92-087 modifies valves 1SI-853A-D, and CV-370 to install a pin in the hanger arm and disc stud, and removes the anti-rotation stubs to allow the disc to fully seat and prevent disc rotation.

Summary of Safety Evaluation: Based on a review of NRC Generic Letter 88-17, "Loss of Decay Heat Removal," and WE responses, there is no violation of our commitments regarding reduced inventory operation provided some precautions are met. CV-370, must be isolated from the cold leg, otherwise this could result in a cold leg opening without an acceptable hot leg vent. The opening of the SI-853A-D check valves results in hot leg openings and are therefore, acceptable. The injection lines from containment are isolable via the motor-operated valves for containment closure concerns.

A minimum of two methods of decay heat removal are available at all times. Both trains of RHR are available (pumps and heat exchangers), which satisfies this requirement. Two methods of reactor coolant system (RCS) makeup are required to be available. This is satisfied by combinations of safety injection (SI) pumps, refueling water storage tank (RWST) connection to the RCS and charging through auxiliary charging. RCS boration is available via CVCS auxiliary charging, and if necessary, safeguards pumps from the RWST.

Commitments to GL 88-17 are incorporated into the existing shutdown procedures. These procedures and the above precautions provide administrative controls to ensure reduced inventory commitments are satisfied. There are no Technical Specification conflicts while working simultaneously all five valves.

A review of the boron precipitation issue indicates a commitment to be able to inject emergency core cooling system (ECCS) water into the hot legs or the upper plenum of the reactor vessel within 14 hours of a large-break LOCA. The purpose of this is to prevent uncovery of the core in the event that boron precipitates and blocks cooling water flow into the bottom of the reactor core via the cold legs. At a low decay heat level and the RCS and associated metal mass is at a cold condition, this postulated phenomena is incredible during the short time the upper plenum injection flow paths are not available.

Only two 853 valves are worked at one time to maintain one upper plenum injection flow path available. (SER 92-055)

72. MRs 92-088*A, *B, *C, *D, *E, *F, *G, and 92-088-01 (Common), Gas Turbine. MR 92-088*A installs a rotor position changer to allow rotation of the turbine shaft on a timed basis; MR 92-088*B upgrades the air supply filtration system; MR 92-008*C installs a lube oil temperature controller retrofit; MR 92-088*D upgrades the glycol cooling system; MR 92-088*E improves the transition seal system; MR 92-088*F performs a compressor inlet airflow measurement; MR 92-088*G provides trunnion lubrication; and MR 92-088-01 replaces the transmission temperature transmitters.

Summary of Safety Evaluation: While G05 is out of service, a temporary diesel generator (TDG) is used as a power source for SBO and Appendix R alternate shutdown scenarios. Installation and siting of the TDG was addressed in SER 92-049.

During the overhaul, G05 is disconnected from tie bus H01 and tagged out along with most of its supporting systems in accordance with the equipment isolation procedure to allow work on these systems. The gas turbine building sprinkler system was tagged out. This action was reviewed and approved by the Fire Protection and Safety Coordinator. The gas turbine building is isolated from the rest of the plant with the exception of its close proximity to the X04 transformers. The overhaul-related activity which could impact the X04 transformers is physical contact with the crane which is brought in to assist in the overhaul. Crane controls are in accordance with internal procedures. The loss of the X04 transformers was previously analyzed.

Post-overhaul testing consists of post-modification testing as applicable, a pre-startup controls and electrical checkout, and monitoring of critical parameters upon startup prior to loading the machine. Operational testing is performed in accordance with established procedures.

The overhauled gas turbine is an enhancement of the existing unit and is not fundamentally different. (SER 92-062)

73. MR 92-092 (Unit 2), Auxiliary Feedwater System. The P29 turbine-driven auxiliary feedwater pump mini-recirculation valve, 2AF-4002 was added to the IST program. This program requires the fail safe test and stroke test close of AF-4002 on an annual basis during cold shutdown. MR 92-092 adds an air line which is routed around AOV control solenoid valve AF-4002-S, and provides instrument air to the air-operated valve (AOV) for the testing.

Summary of Safety Evaluation: The Whitey 2-way ball valve installed by MR 92-092 is used to control the flow of instrument air (IA) to and from the AOV during testing (opening and closing the AOV). The ball valve and its position is labeled and during normal operations the ball valve is locked in its required position.

MR 92-092 does not increase the probability of instrument air failure to the AOV. Should the new installation fail, the consequences are no different than the original design; the valve would fail shut.

Installation does not change the normal operation of AF-4002, while it allows the required testing to be performed. Its failure causes the AOV to go to its original design for fail safe position, is evident at the control boards or locally, and can be dealt with by use of the AOV gag and by the use of electric auxiliary feedwater pumps as appropriate and as required. There are no adverse system interactions. (SER 92-076)

74. MR 92-094 (Common), Emergency Diesel Generator. MR 92-094 provides new cabinet base anchorage for diesel control cabinets C34 and C35.

Summary of Safety Evaluation: This change is required as C34 and C35 contain relays which perform safety-related control functions for diesel generator starting and control including generator field. Diesel control cabinets C34 and C35 are located in the control building, diesel generator rooms on El. 8' between column lines A - C and 10 - 12. These cabinets are modified as a result of condition report 92-371.

Diesel control cabinets C34 and C35 and diesel generators G01 and G02 shall not be taken out of service deenergized. Although the equipment within these cabinets is important to safety, precautions are taken to prevent disabling of the affected equipment. Operation of diesel generators G01 and G02 and equipment associated with cabinets C34 and C35 is verified in accordance with TS tests. Precautions are taken to prevent any metal filings or debris from falling into or on the equipment within the affected cabinets. Non-conductive tools are utilized for work inside the diesel control cabinets. Therefore, operability of diesel control cabinets C34 and C35, diesel generators G01 and G02, and any equipment important to safety in the location of the upgrade is unaffected. (SER 92-057)

75. MR 92-106 (Unit 2), Containment Piping Supports. MR 92-106 upgrades six existing piping supports inside the Unit 2 containment structure to meet seismic analysis requirements.

Summary of Safety Evaluation: MR 92-106 is performed during a refueling outage. During this time, active performance of the pressurizer spray, the component cooling to the reactor coolant pump and the accumulator systems is not required; therefore, an LCO is not entered. Piping restraint AC-152N-2-SB2 is disconnected while being modified. It is a small bore piping restraint of a riser with only nominal seismic loads. AC-152N-2-SB2 is disconnected as long as necessary to refabricate and reattach the restraint. There is no significant masses on or near the adjacent spans and the spans are of reasonable lengths. The adjacent supports, AC-152N-2-SB1 and AC-152N-2-SB3, are capable of adequately restraining the loads (using short-term operability limits) during a seismic event. Based on this evaluation, a temporary restraint is not required

for AC-152N-2-SB2. Other supports upgraded do not require disconnecting the piping support from the building. Therefore, no piping supports modified require temporary support. The piping system meets applicable Code requirements for piping and supports. (SER 92-075)

76. MR 92-106-01 (Unit 2), Residual Heat Removal System. MR 92-106-01 upgrades one existing piping support inside Unit 2 containment to meet analysis requirements and removes one support to allow free thermal expansion which meets commitments made during IEB 79-14 as-built reconciliation program. The upgrade consists of replacing the existing non-standard riser clamp with a standard heavy-duty one.

Summary of Safety Evaluation: The as-built piping and support for the residual heat removal (RHR) system as evaluated for its ability to withstand the design basis loads and stresses imposed on the system. The results of the analysis show, while the system is operable as is, the pipe clamp for AC-2501R-2-AC2 requires replacement. The as-built piping and piping supports and the modification are documented in ABR 14-19. The piping support upgrade includes replacement of the riser clamp. Temporary support of the piping shall be provided and the piping system shall remain operable throughout the clamp replacement.

The as-built piping and supports for the charging system were evaluated for their ability to withstand the design basis loads and stresses imposed on the system. The as-built and the modified condition acceptability are documented in stress and support reports 200070 and 700070S. The analyses also document the operability of the system. The results of the analyses show the system is overloaded due to thermal expansion of the piping and the overload can be eliminated by the removal of one piping support (CH-2502R-3-2S260). (SER 92-100)

Summary of Safety Evaluation: Further engineering analysis documented that CH-2502R-3-25260 may remain in place. This is documented in support report 200070S. (SER 92-100-01)

77. MR 92-107 (Common), Gas Turbine. MR 92-107 installs an enhanced vibration monitoring system.

Summary of Safety Evaluation: While G05 is out of service, a temporary diesel generator (TDG) is used as a power source for SBO and Appendix R alternate shutdown scenarios. Installation and siting of the TDG was addressed in SER 92-049.

During the overhaul ..05 is disconnected from tie bus H01 and tagged out along with most of its supporting systems in accordance with the equipment isolation procedure to allow work on these systems. The gas turbine building sprinkler system was tagged out. This action was reviewed and approved by the Fire Protection and Safety Coordinator. The gas turbine building is isolated from the rest of the plant with the exception of its close proximity to the X04 transformers. The overhaul-related activity which could impact the X04 transformers is physical contact with the crane which is brought in to assist in the overhaul. Crane controls are in accordance with internal procedures. The loss of the X04 transformers was previously analyzed.

Post-overhaul testing consists of post-modification testing as applicable, a pre-startup controls and electrical checkout, and monitoring of critical parameters upon startup prior to loading the machine. Operational testing is performed in accordance with established procedures.

The overhauled gas turbine is an enhancement of the existing unit and is not fundamentally different. (SER 92-062)

78. MR 92-122 (Unit 1), IWP 92-122, CR 92-360, Instrumentation. MR 92-122 installs new anchorages for the plant process and control cabinets 1C-105 through 1C-114 and 1C-115 through 1C-133 located on El. 44' of the control building in the southwest corner of the control room.

Summary of Safety Evaluation: The new anchorage detail is installed at the base of each cabinet row, on the north and south sides of each cabinet row, in the east/west direction the long axis of the cabinets. The anchorage detail is made of angle steel secured to the cabinets and to the control room floor. The anchorages are designed to withstand the SSE at the control building El. 44'.

The cabinets worked on include the analog protection instrumentation (1C-111 through 1C-118) which provide input to the safeguards relay racks, the in-core instrumentation (1C-121 through 1C-124), safety injection (SI) and auxiliary coolant control panel (1C-129), the nuclear instrumentation (1C-130 through 1C-133). These systems are used in response to accidents evaluated in the FSAR. The cabinets of most concern are the ones associated with the four channels of reactor protection (1C-111 through 1C-118) and the four channels of nuclear instrumentation (1C-130 through 1C-133). The work on the cabinets is performed at the base of the cabinets out of the vicinity of the instrumentation. The installation work plan also requires I&C personnel to be present during the work to ensure precautions are taken to prevent impact on the instruments. The drilling required during the installation produces some vibration in the cabinets. Personnel are present to provide continuous visual monitoring for adverse effects resulting from vibration. If a potential for any adverse effects is found, the work is halted and an evaluation is conducted to determine whether to continue the installation at the present plant conditions. Therefore, the systems controlled by this instrumentation remain available during installation. The systems are available for response to an accident as evaluated in the FSAR.

The reactor protection, engineered safeguards, and process instrumentation in the affected instrument racks were declared inoperable because the cabinets containing the instruments inadequately mounted to withstand the design basis safe shutdown earthquake. The instruments remain operable and function for all operating conditions other than a seismic event. The design basis does not consider a seismic event an initiator of any design basis accidents in the FSAR. Therefore, under normal operating conditions reactor protection and engineered safeguards systems are capable of performing their function.

The installation work plan ensures administrative controls are in place to minimize risks associated with work performed in the cabinets. (SER 92-065)

79. MR 92-144*A (Common), IWP 92-144*A, Component Cooling System. MR 92-144*A installs the mechanical scope, which includes the CC-LW-63 and CC-LW-64 valve replacement and piping rework to add seat leakrate testing capabilities.

Summary of Safety Evaluation: The modification on the component cooling (CC) system supply: a.) Replaces the existing 6" butterfly valve (LW-63) with a dimensionally similar leaktight, 6" butterfly valve; b.) Installs a 6" manual gate valve (CC-695) on the vertical riser downstream of isolation valve LW-63; c.) Installs a 3/4" Sch. 80, pressure tap line with shutoff valve (CC-694) on the vertical riser between valves LW-63 and CC-695; d.) Raises the supply to the cryogenic gas compressors and aftercoolers to the vertical run downstream of valve CC-695; e.) Removes local pressure gauge (PI-LW38) along with the gauge isolation valve CC-828, and sensing line; and f.) Installs a 3/4" Sch. 80, test connection with shutoff valve (CC-693) on the vertical riser between LW-63 and CC-826.

The modification on the CC system return: a.) Replaces the existing 6" butterfly valve (LW-64) with a dimensionally similar, leaktight, 6" butterfly valve; b.) Installs a 6" manual gate valve (CC-699) on the vertical riser upstream of isolation valve LW-64; c.) Installs a 3/4" Sch. 80, pressure tap line with shutoff valve (CC-698) on the vertical riser between valves CC-699 and LW-64; d.) Raises the return of the cryogenic gas compressors and aftercoolers to the vertical run upstream of valve CC-699; e.) Installs a 3/4" Sch. 80, pressure tap line with shutoff valve (CC-697) on the vertical riser just upstream of gate valve CC-696; f.) Installs a 3/4" Sch. 80, high point drain line with shutoff valve (CC-696) on the horizontal run downstream of valve LW-64; and g.) Removes local pressure gauge (PI-LW39) along with the gauge isolation valve CC-891, and sensing line.

FSAR Appendix A requires the interface between the Class 1 and Class 3 portions of the CC system be provided with the capability of remote operation from the control room or be normally shut. The CC system must meet the requirements of a closed system outside containment which provides the secondary containment isolation boundary.

MR 92-144*A enhances the performance, increases the reliability and ensures the conformance to design criteria of an existing system. This modification does not create any new radiological release scenarios or failure modes different from the original design during either construction or operation. The portion of the CC system affected by the modification, downstream of the isolation valves, has no safety-related function. Systems affected by the radwaste CC system were placed out of service prior to performing the modification to protect against equipment damage. During replacement of the valves, the main CC system was able to perform its intended design function since it was isolated by the red tagged shut gate valves. <u>(SER 92-094)</u>

80. <u>MR 92-144*B (Common), IWP 92-144*B, MWR 92-5100 and 92-5101, and Augmented MWR 5440</u>, Component Cooling Water System. MR 92-144*B installs the electrical scope, which includes installation of control switches and indicating lights in the control room for CC-LW-63 and CC-LW-64 valves.

<u>Summary of Safety Evaluation</u>: The electrical scope of work required to accomplish the modifications described in MR 92-144*B includes: a.) Disconnection and connection of new control wiring at the valves, located on PAB El. 8'; b.) Disconnection and removal of the existing control switches and indicating lights on panel C180 on PAB El. 26'; c.) Installation and connection of new fuses, new control switches and indicating lights, with appropriate nameplates, in control room panel 2C03; d.) Installation of interpanel wiring to connect the containment isolation (CI) contacts from auxiliary relays 2CI12-X1 and 2CI21-X1, located inside the transition piece between C01 and C02 in the control room to the new control circuit; e.) Installation of a new terminal box TB-166 near the existing terminal box for the wiring associated with the solenoid valves and position switches on PAB El. 8'. The new terminal box allows future interlocks of valve closure for radwaste equipment protection through an interposing relay which is installed in the box; f.) Installation of a new safety-related relay for CI signal interface. Two new cables from panel 2C03 to terminal box TB-166 were installed as part of augmented MWR 5440 in "A" train cable tray at El. 8' near the valves. The final connection is made as part of the modification using field routed seismic conduit; g.) Installation of wires in 2C03 from the annunciator to riser #54 for future connection of the annunciator circuit; and f.) Installation of jumpers in 2C03 to deactivate the automatic CI closure of these valves.

The control circuit (completed under MR 92-144*C) provides for automatic closure of valve CC-LW-63 and CC-LW-64 on a CI signal from either train. The CI inputs are taken from existing spare contacts on CI relays 2CI12-X1 and 2CI21-X1, located inside the transition piece between C01 and C02 in the control room. They are both electrically isolated to enable them to be treated as "A" train internal to the control cabinet and are routed as such internally. The circuit operating power is obtained from the current

source of safety-related DC power to panel 2C03 (D17 breaker 8) via BUSS 2.5 amp FRN-R fuses to ensure proper coordination with the sup.ply breaker. The cable from the control room to the valve operators and the new wiring from the CI relays (in C01/C02) is routed as electrical "A" train.

The CC system must meet the requirements of a closed system outside containment which provides the secondary containment boundary. The existing control switches are relocated to the main control room to resolve the FSAR Appendix A concerns. The control circuits for the isolation valves are upgraded from non-safety-related to safety-related to satisfy WE regulatory commitment to ensure that the CC system remains a closed loop system.

MR 92-144*B enhances the ability of the control room operators to more effectively mitig *e the consequences of a malfunction in the CC system outside containment. (SER 92-094-01)

TEMPORARY MODIFICATIONS

1. TMs 92-007 (Unit 1), and 92-009 (Unit 2), Service Water System. TMs 92-007 and 92-009 installs blank flanges upstream of TCV-12A and TCV-12D on the service water outlet side of the component cooling heat exchangers in the bypass line around the 12" main outlet piping.

   Summary of Safety Evaluation: The service water outlet bypass lines from component cooling heat exch[...]gers 1HX-12A and 2HX-12D are isolated by installing a blank orifice flange on the [u]pstream side of the temperature control valves. Isolation of the lines is required to replace the pitted and eroded piping downstream of the temperature control valves.

   The piping in which the blanks are installed is Class JB, which specifies a 150# ASA ASTM A-181 flange. A 150# stainless steel Type 304 blank orifice flan[g]e with Flexitallic gaskets on either side is used. The use of stainless steel Type 304 flanges in the service water system was evaluated and found acceptable during MR 87-158. A non-QA flange is dedicated for use in the QA system. The blank flange is given a restricted QA release number (RQAR).

   Stress calculation was performed to determine the effect of an unsupported TCV (if downstream pipe breaks) on the stub pipe connection to the 12" service water pipe. The calculation determined the valve did not have to be supported to ensure integrity of this pipe. (SER 92-014)

2. TM 92-012 (Unit 1), Rod Position Indication System. TM 92-012 switches the rod position indication (RPI) signal cables for control rods at core locations C-7 and G-11 in control Bank D at the inputs to the RPI signal conditioning cards.

   Summary of Safety Evaluation: The signal cables can later be swapped at the patch panel located on the reactor vessel head.

   Positions for control rods C-7 and G-11 in Bank D initially indicate on the displays for the opposite rod (i.e., C-7 indicates on the meter labeled "G-11" and G-11 indicates on the meter labeled "C-7"). Temporary information tags or stickers at each location where control position is displayed alerts operators to the off-normal condition. Later, rod indication is restored such tha[t] each rod indicates on the proper display. (SER 92-030)

3. TM 92-014 (Unit 1), Chemical [...]me Control System. TM 92-014 temporarily replaces the impulse tubing [...] Unit 1 letdown line flow transmitter 1FT-134, with flexible hoses to facilitate ins[...]on of the Unit 1 containment spray full flow test line under MR 88-097*A.

   Summary of Safety Evaluation: The impulse lines for 1FT-134 interfere with welding and grinding work for the containment spray tie-ins. TM 92-014 was installed during U1R19 and removed prior to leaving cold shutdown.

   The temporary use of braided flexible stainless steel hoses for the impulse lines does not affect the operation of the flow transmitter. The pressure and temperature ratings of the hoses are in excess of that required for the application and the materials are suitable for contact with reactor coolant. A pre-service and initial service check ensures the temporary impulse lines are leak tight. The hoses are sloped to avoid air pockets in the lines.

Operability of the flow transmitter is not required by Technical Specifications. If a leak developed in the impulse lines, it could be isolated by shutting the transmitter's root valves which allows the letdown line to remain in service. During installation and removal of TM 92-014, the transmitter is isolated by the root valves. This removes the letdown line flow indication from service for a short period of time (approximately 1 hour). This flow indication indicates the amount of water removed from the reactor coolant system (RCS) via the letdown line and is used to monitor inventory changes. The RCS is at mid-loop conditions when this temporary modification is installed. At this time the charging and letdown systems are lined up so the volume control tank is bypassed. With this lineup, RCS inventory is maintained constant and charging flow indication can be used to determine CVCS flow. In addition, the reduced inventory level indicators are available and are not affected by this temporary modification. (SER 92-037)

4. TM 92-017 (Unit 1), Chemical and Volume Control System. TM 92-017 installs lead shielding on portions of the regenerative heat exchanger, auxiliary spray valve CV-296, and the charging isolation to reactor coolant system (RCS) loop "A" valve CV-1298.

Summary of Safety Evaluation: Radiation levels from the regenerative heat exchanger (RHE) are very high. Temporary lead shielding is placed on the RHE, around CV-1298 and CV-296 to permit workers to perform weld examinations on the RHE without receiving excessive dose. The TM permits placing lead shielding on one valve and one side of the RHE at one time to prevent excessive loading. The lead shielding is removed after the examinations are complete. The shielding is in place only when the reactor is in a cold or refueling shutdown condition.

The lead shielding was evaluated because the proposed activity could affect the function or method of function of the regenerative heat exchangers, letdown piping, charging, and auxiliary spray systems. The systems are Seismic Class 1. None of the systems are required for accident mitigation at shutdown conditions. None of the valves in the regenerative heat exchanger cubicle are containment isolation valves, therefore, the proposed activity does not affect containment integrity.

The weight of the lead blankets does not exceed the design deadweight loading of the RHE or the piping associated with valves CV-1298 and CV-296. However, the presence of the lead blankets introduces a non-seismically supported load. Therefore, the use of lead blankets is restricted to when the reactor is shut down and the system in the RHE cubicle does not perform a safety function.

Failure of the 2" letdown line in the RHE cubicle coincident with the failure of MOV-427, which is outside of the cubicle, results in an uncontrollable loss of reactor coolant. In this event, the break would be isolable via the manual valve, RC-558. Letdown capability is still available via excess letdown or residual heat removal (RHR).

Failure of the 3" charging line coincident with the failure of check valve RC-295, which is outside the cubicle, results in an uncontrollable loss of reactor coolant. MOV-1298 is in the RHE cubicle and the break is assumed to be upstream of it. This break is not isolable and results in a sustained small break LOCA. Charging would no longer be available through the normal route, but the capability for this still exists via auxiliary charging, and RHR or safety injection when water is available from the refueling water storage tank.

A failure of auxiliary spray 2" line off the charging line (assumed upstream of CV-296 and check valve 297) would at worst depressurize the reactor coolant system (RCS). With the pressurizer either solid or depressurized, auxiliary spray has no direct safety function. Little or no actual inventory would be lost because the spray line comes in at

the top of the pressurizer. A failure downstream of these valves would need to be coincident with the failure of RC-295 check valve and MOV-1298. This is the same scenario as the charging line failure but on a smaller line.

A failure of any or all three of these systems during a plant shutdown is not a major safety concern. None of the systems need to be operable for safety reasons. Due to the piping arrangements and the location of the RCS penetrations above the RCS piping centerline, RCS inventory cannot be reduced below the mid-loop level. With the cavity flooded and only one train of RHR operational, leakage can be made up by auxiliary charging. Leakage indication is available via reactor vessel level and/or sump A level changes. Required safety functions, (water to keep the core covered, boration paths, heat removal, etc.) are still available via safety systems. (SER 92-046)

5. TM 92-018 (Unit 1), Safety Injection System. TM 92-018 installs a stem clamp on 1SI-856B in the closed position. The stem clamp is required since the "B" train residual heat removal (RHR) pump is to be run while the valve operator is removed from the valve.

Summary of Safety Evaluation: Valve 1SI-856B is normally kept in the shut position during the phase of refueling when the cavity is flooded for fuel motion. The valve provides a redundant isolation for check valve 1SI-854B. The valves isolate the empty refueling water storage tank from the suction piping of "B" train RHR pump P10B. 1SI-856B has no automatic functions.

Clamping the valve in the shut position ensures the valve does not open if the check valve leaks while the valve operator is removed for preventive maintenance.

There are no seismic concerns since the valve stem clamp weighs less than the valve operator. The lower weight reduces seismic stresses. The stem is not unsupported, but the seismic loading on the approximate 24" length is minimal. (SER 92-043)

6. TM 92-020, 480 V Electrical System. A temporary diesel generator (TDG) is used as a standby power supply for the alternate shutdown system during the G05 combustion turbine maintenance outage.

Summary of Safety Evaluation: Concerns associated with this temporary modification are the siting hazards (i.e. fire protection), Appendix R alternate shutdown capability for a postulated fire in the 4160 V vital switchgear room, and station blackout (SBO).

The siting concerns are met in part through meeting FPER requirements. The trailer on which the diesel sits has a mechanism for containing a fuel oil spill. This is primarily an environmental concern, as there is a storm drain within proximity of the temporary diesel generator. A fire attack plan for the TDG was issued to the fire brigade chiefs. The new cable routing and connection do not violate any existing fire zone requirements. Possible missiles from the TDG do not pose an increased threat to equipment important to safety.

During the G05 combustion turbine maintenance period, the TDG provides AC power to the B09 alternate shutdown bus in the event of a SBO. The TDG in this capacity is able to power the safe shutdown equipment as described in Appendix R. Given a SBO, the plant is able to achieve safe shutdown. (SER 92-049)

7. TM 92-029 (Common), 480 V Electrical System. TM 92-029 provides a standby emergency power supply for the technical support center (TSC) while G501 auxiliary diesel is out-of-service for repair. The power supply is a temporary gas turbine (TGT). Backup power for the TSC is required by NUREG-0696 and NRC Confirmatory Order.

Summary of Safety Evaluation: The temporary gas turbine (TGT) has sufficient capacity to provide TSC load requirements. The TGTs power output cables connect into the existing power supply cables on the load side of breaker 52T and are tested prior to being put into service. The machine has a 300 gallon fuel supply which is sufficient for 12 hours of operation. Arrangements were made to bring in additional kerosene as required.

The TGT is started by a manual start button located at the unit which is different than the automatic start feature of G501. The TGT is fully dedicated to supplying backup TSC power. The small amount of time required to manually start the TGT does not affect the overall performance of the TSC. (SER 92-061)

8. TM 92-038 (Common), Chemical and Volume Control System. TM 92-038 disconnects the grounded heater and allows the remaining heaters in T6B boric acid storage tank (BAST) to operate. This is required to maintain tank temperature high sufficient for the 12% boric acid solution in T6B to remain fluid.

Summary of Safety Evaluation: The T6B heaters are not required to be operable because the matter of fluidity is not in question for a BAST which is not in service. The remaining heaters are maintained operable in order to maintain fluidity.

If the remaining heaters are capable of maintaining T6B temperature >145°F, then placing T6B in service does not involve an unreviewed safety question since the requirement of TS 15.3.2.C.3 is that the tank temperature be maintained at >145°F. The ability of the remaining heaters to maintain T6B at >145°F is verified prior to placing T6B into service. (SER 92-064)

9. TM 92-047 (Unit 2), Main Steam System. TM 92-047 installs and removes an inflatable cylindrical stopper to act as a vapor barrier to maintain containment integrity during "B" steam generator tube plugging and fuel movement.

Summary of Safety Evaluation: The main steam line is considered a potential vent path during the tube plugging work on Unit 2 "B" steam generator because the primary side manways are opened, and the leak hole in the steam generator tubing allows a vent path through the steam system. This temporary barrier is required because of work on main steam isolation valve (MSIV) 2MS-2017. The MSIV work has opened and disassembled the valve, and is therefore, not available as a containment barrier. The inflatable plug completely fills the 30" steam piping and holds back up to 60% of the pressure used to fill the bag. The bag is pressurized to ~3 psig, and seals against a differential pressure of 1.8 psig. This is acceptable because of the low differential pressure expected during fuel movement. The only source of containment pressure is the containment ventilation system. The purge supply fan, which supplies air to the containment, is rated for 4" of static water pressure, or 0.17 psig. The inflatable barrier is used to maintain a vapor barrier while tube plugging work is performed. After the completion of tube plugging, the temporary barrier may be removed.

The seal is not associated with safety-related equipment. The malfunction which requires the seal to act as a containment barrier is a malfunction during fuel movement. This malfunction does not increase the pressure inside containment and the seal is adequate as a barrier. The consequences of a malfunction of equipment during the fuel movement would therefore, not be increased by the seal. The release of fission products to the environment does not consider the effect of containment isolation therefore, the seal increases the amount of protection against this accident. (SER 92-095)

10. TM 92-048 (Common). Service Water System. SW-660 had a small pin hole leak due to cavitation damage. TM 92-048 removes valve SW-660 and replaces it with two blind flanges. After installation, flow can only be through SW-661, instead of both, or either SW-660 and SW-661. TM 92-048 is removed when a replacement valve and fittings are available.

Summary of Safety Evaluation: The removal of SW-660 was evaluated per calculation N-92-092 regarding the seismic capability of the modified piping arrangement. The calculation addressed the stresses seen by the damaged reducer fitting (not replaced at this time), regarding its pressure retaining capability. The stresses found are within Code allowables.

The original installation code for this piping was ASME Section 3 Class 2, 1974 Edition, Summer 75 Addenda. The flanges installed, are per Section 3 Class 3, 86 Edition, 87 Addenda. The Code year difference is acceptable as reconciled in ANSI/ASME Code Reconciliation For Replacement Material, Parts and Components, by Reedy Associates, July 1, 1992.

Flow requirements for service water to the spent fuel heat exchangers were evaluated and the effect that reliance on SW-661 will have, i.e., SW-660 removed. Relying on SW-661 is no problem for current conditions but it may be necessary to have SW-660 back in service prior to a core offload. A core offload is scheduled for U1R20. If an unplanned core offload occurred prior to replacement of valve SW-661, the blind flanges could be replaced with a pipe spoolpiece welded to weld neck flanges to provide a flow path for additional service water.

In order to remove SW-660, service water to the spent fuel pool (SFP) heat exchangers is secured. This condition should be limited to keep the temperature rise of the spent fuel pool to a minimum. It is estimated that isolation, draining, valve removal, and reinitiation of flow will take 4 to 6 hours. The normal temperature of the spent fuel pool is maintained between 70 and 90°F. An administrative maximum is 120°F, which is also the temperature alarm setpoint. Based on conservative data obtained from MR M-278, OP-8A states that is takes over 30 hours to heat up the spent fuel pool from 120 to 200°F with cooling secured. This assumes it is done at a time between refueling outages and not during a complete core unload. The evolution would be started with the spent fuel pool temperature at the low end of the normal operating temperature band. In addition, data from IWP 90-056B implies that the temperature rise was <1°F/hr when service water was secured for installation of thermowells for heat exchanger monitoring. This was done approximately 2 months after a refueling outage. The spent fuel pool temperature needs to be maintained below 180°F to ensure that no structural concerns exist. With known data from past experience and the simplicity of the job to be performed, removal of the flanged valve and blind flanging of the openings, ample time exists for the evolution. (SER 92-103)

SPARE PARTS EQUIVALENCY EVALUATION DOCUMENTS (SPEEDs)

1. SPEED 91-087. New Disc Material on Aloyco Gate Valve. The SPEED replaces Aloyco valve original disc material.

Summary of Safety Evaluation: The original valve disc was made of ASTM A296 CF8 contrary to FSAR Section 6.2-21. The new disc was manufactured by the original equipment manufacturer using the same design and drawings. The dimensions and pressure retaining capabilities remain the same. The new disc is made of ASME SA 479 TP304 and is equal to or better than the original disc design. The disc is made of

austenitic solution annealed stainless steel with the same corrosion resistance characteristics. The change in material from cast to wrought iron does not affect the disc or valve from performing its safety-related function. Because of the change in form, ultrasonic examination is the preferred method of volumetric examination and achieves equal results as radiography did on cast form. Surface examination was performed by liquid penetrant testing. (SER 92-006)

2. SPEED 91-086, New Disc Material on Aloyco Manual Gate Valve. The SPEED replaces Aloyco original disc material.

   Summary of Safety Evaluation: The original valve disc was made of ASTM 296 CF8M contrary to FSAR Section 6.2-21. The new disc is manufactured by the original equipment manufacturer using the same design and drawings. The dimensions and pressure retaining capabilities remain the same. The new disc is made of ASME SA 479 TP316 and is equal to or better than the original disc design. There is no change in fit or function. The disc is made from austenitic solution annealed stainless steel with the same corrosion resistance characteristics. The change in material from cast to wrought does not affect the disc or valve from performing its safety-related function. Because of the change in form, ultrasonic examination is the preferred method of volumetric examination and achieves equal results as radiography did on cast form. Surface examination was performed by liquid penetrant testing. (SER 92-008)

3. SPEED 92-012, Replacement of Solenoid Valves 1(2)CV-112A-S. The SPEED replaces solenoid valves 1(2)CV-112A with another solenoid valve which is equivalent except it cannot be manually operated.

   Summary of Safety Evaluation: CV-112A is a 3-way, air-operated valve (AOV) which diverts letdown flow from the volume control tank to the holdup tanks. Two solenoid valves are provided for control of this AOV. According to a previous analysis, (SPEED 88-082) a replacement valve was determined to be equivalent to one of the solenoid valves for this AOV. The other solenoid valve is equivalent to the other except that it has a manual operator. SPEED 92-012 provides additional justification necessary to allow the manual operator to be removed from the second solenoid valve in order to allow replacement according to the analysis provided by SPEED 88-082.

   Removal of the manual operator from the solenoid valve does not affect the operation of the valve nor the air-operated valve it controls. (SER 92-015)

4. SPEED 92-023, Shaft Material Upgrade for Main Steam Isolation Valves (MSIVs). The original shaft material was susceptible to cracking in the keyways if not properly heat tested.

   Summary of Safety Evaluation: SPEED 92-023 upgrades MSIV valve shaft material from Type ASTM A-276 to ASTM 564 GR 630 COND 115om. The original material could have cracked in the keyway areas if the valves were improperly heat tested. The original shafts were noted as not showing any stress cracking as observed per the 10 CFR 21 report. The new material assures component integrity and reduces the probability of valve malfunction. (SER 92-035)

5. SPEED 92-082, Removal of Unit 2 Steam Generator Tube Lane Blocks. The SPEED permanently removes the Unit 2 steam generator tube lane blocks. Removal is consistent with normal maintenance (sludge lancing) performance during each outage.

   Summary of Safety Evaluation: The Unit 2 tube lane blocks are not part of the original steam generators. Tube lane blocks were installed in an effort to increase flow across the tubesheet area to effect a sweeping of the tubesheet, thus reducing the amount of

sludge accumulation. Sludge accumulation in low flow areas is considered to be a prime contributor to tube degradation.

A quantitative analysis of the thermal-hydraulic benefit of the tube lane blocks is not available. However, it was concluded that ~100 more tubes would move into the potential sludge deposition zone with removal of the tube lane blocks. For Series 44 steam generators, approximately one-half of the tube bundle could be subject to sludge deposition at the tubesheet. Therefore, the additional 100 tubes would account for a 6% increase if the blocks were removed. Regular sludge lancing reduces the impact of the additional sludge deposition thus reducing tube degradation. If so, the 6% increase is expected to have a negligible impact on tube degradation.

Although the tube lane blocks can have a local effect on the hydraulic conditions at the tubesheet, they have no impact on the heat transfer conditions. Thus, they have no effect on steam pressure or flow. (SER 92-090)

5. SPEEDs 92-101, 92-102, and 92-103, Material Change of Valve Internals for Valves 1&2SI-853A-D and 1&2SI-889A&B. The SPEEDs document material changes recommended by the valve manufacturer.

Summary of Safety Evaluation: The referred swing check valves have a history of sticking in the open position. In Fall, 1991, and Spring, 1992, low head safety injection valves 1SI-853D and 2SI-853D did not fully shut after cycling during Event V testing. Radiography of these valves showed the disc was approximately 1/4" to 1/2" off of the bottom valve seating surface. The radiography did not indicate obstructions or valve damage. Upon inspection of the Unit 1 valve, it was discovered that friction between the valve hanger arm anti-rotation stub and the disc anti-rotation stub may have prevented it from fully engaging the seat. Further root cause analysis revealed this phenomenon was described in previous operating experience documents (SER 20-83, NRC IN 83-06). Based upon this information, the Unit 1 valves were to be modified in accordance with the valve manufacturer's recommendations by installing a pin in the hanger arm and disc stud, and removal of the anti-rotation stubs (or rotate the disc 180°) to allow the disc to fully seat and prevent disc rotation. The manufacturer has redesigned the internal parts for these valves. The new parts correct the potential sticking open problems.

The scope of the new parts change included 2SI-853A, B, C, and D. Valves 2CV-295 and 2CV-370 are similar but are addressed separately. 1&2SI-889A&B are similar valves, but are not scheduled for this modification. (SER 92-098)

MISCELLANEOUS EVALUATIONS

1. Condition Report 92-613 (Unit 2), Debris in Safety Injection, Containment Spray, and Refueling Water Storage Tank. Debris was discovered in and subsequently removed from portions of the Unit 2 safety injection (SI) and containment spray (CS) systems (including "A" train suction supply from residual heat removal (RHR)), and the Unit 2 refueling water storage tank (RWST). Most debris was small and consisted of fragments of tape, foam rubber, and other "soft" debris, along with metallic debris.

The debris was initially discovered when the "A" train CS pump failed its quarterly test because a foam rubber plug blocked the pump suction. LER 92-003-01 addressed this event and provided a safety assessment. Inspections were performed during U2R18 of portions of the Unit 2 CS, RHR, and SI systems to identify additional foreign material in these systems. The inspections included, to the extent practicable, the portions of the

systems affected by the full-flow test line modification (MR 88-098; a probable source of the foam rubber plug). The location of the inspections included likely trapping points in the full-flow test lines, likely trapping points in the mini-recirc lines, the RWST, common suction piping from the RWST, and selected portions of dead legs in the CS discharge lines and RHR suction lines. The inspections were performed using a combination of visual inspection, borescopic examination, and radiography. Radiography, with water present in the pipe, was able to detect a ball of duct tape of ≈1" diameter in a test, and any metallic objects of significant size. Borescopic examination detects objects as small as 1/8-1/4", provided they are within the field of view.

The Unit 2 RWST interior was inspected using a remote controlled minisub and video camera and by personnel entry. Debris found included small pieces of tape, Herculite, and other material. As a result of the personnel entry, all RWST debris was removed.

Summary of Safety Evaluation: This SER is based upon and incorporates SER 86-028 and SER 88-134 which includes the NSSS supplier's safety evaluation for the split pin remnant and similar remnants potentially present in the reactor coolant system (RCS), chemical and volume control system (CVCS), and residual heat removal (RHR). SER 88-134 documents a missing nozzle dam flange ring insert. These SERs found no unreviewed safety questions associated with metallic loose parts potentially present in these systems. The debris currently found is comparable to that previously evaluated. This SER also includes the safety evaluation present in LER 92-003-01 which directly pertains to safety concerns generated by this debris. Consideration was given to possible cumulative effects of debris in the RCS. Based on the inspection and cleaning performed, the continuing operability tests performed on systems connected to the reactor coolant system, inservice inspection vessel inspections insurance, the debris filter bottom nozzle and failed fuel detection, and the potential impact of any additional debris deposited is minimal.

The thoroughness of the inspections, clean-up, and operability testing; the nature of the debris found; continued system tests per Technical Specifications; the findings of SER 86-028 and SER 88-134; and the safety evaluation in LER 92-003-01 determines the finding of this SER that "accepting as-is" the affected systems, does not involve an unreviewed safety question. (SER 92-105)

2. ECR NE 92-047, D105 PAB Battery Room "A" Air Duct. MR 87-156*A installs an air duct heater in the ventilation supply for D-105 PAB battery room "A". The heater cannot be mounted as originally designed and requires a section of the duct be replaced before installation.

Summary of Safety Evaluation: ECR NE 92-047 removes a 12" flanged section of the duct and cuts 9" off the adjoining section upstream. The end of the upstream section that was cut has a prefabricated slip-on flange attached to provide a means of attaching the replacement section. The prefabricated replacement section is then installed and the heater mounted. The replacement section of the duct has the same internal area dimensions as the sections replaced.

The location of the replacement section is downstream of DMP 5, the air supply damper for battery room "A". This allows the alteration to take place without impacting ventilation to the other rooms served by the HVAC system. The normal ventilation supply to battery room "A" is removed during the installation and temporary ventilation arranged if necessary during the installation.

The D105 battery continues to operate normally both during and after the duct section installation. After installation of the duct section, the PAB battery and inverter room HVAC system operates normally and its pressure rating (10" water) is maintained. The replacement does not affect the operation of the battery room's fire damper during or after installation. (SER 92-024)

3. Fuel Assembly X-19 Extraction Work Plan. The work plan utilizes a 2-ton chainfall between the spent fuel pool (SFP) hoist and the fuel handling tool to allow better control during hoisting and removal of fuel assembly X-19 from location SJ-37.

Summary of Safety Evaluation: This work plan reduces the probability of damaging the fuel assembly when extraction occurs from the suspect SFP locations. The 2-ton load rating provides sufficient margin to the maximum expected load of ~1500 pounds. The chainfall is load tested prior to use. The overload indication and protection remains in service on the SFP crane, precluding the possibility of overloading the hoist or crane. (SER 92-041)

4. LERs 90-010 and 92-001, HPES 90-002, PPCS Axial Flux Difference (Delta Flux) Update Frequency and Technical Specification Bases Change. The plant process computer system (PPCS) calculation for axial flux difference (AFD) changed from one-minute average NIS readings and updating once per minute to using current NIS readings and updating once every 4 seconds. The Technical Specification Bases for the axial flux difference alarm changed to be more general in the description of the actual implementation of the axial flux difference alarm on the plant process computer.

Summary of Safety Evaluation: Computer hardware connections to the plant remain unchanged. The change to the bases does not alter the intent of the Technical Specification regarding the axial flux difference alarm. The specification states that alarms shall normally be used to indicate nonconformance with AFD requirements. Since the AFD alarm remains on the plant process computer, the specification continues to be satisfied.

The TS does not state how the AFD alarm on the computer operates, or that the AFD alarm must be on the computer. The change to the Bases is needed, however, to reconcile the bases to the revised software. (SER 92-070)

5. Temporary Pump Effluent Sump to Ditch. The bypass of the effluent sump around the normal retention pond effluent point, straight into a ditch which drains into Lake Michigan was evaluated. Both effluent sump pumps were in need of repair, making this action necessary until the pumps were repaired. This alignment bypasses the retention pond 30-day retention and RMS effluent monitor.

Summary of Safety Evaluation: The implications with Technical Specification requirements are clear and specific. The implications with the FSAR deal with the consequences of the loss of the decay time in the retention pond. Since normal plant releases through this release point are well below regulatory limits, the consequences of this loss are negligible.

Sampling the effluent stream in accordance with Technical Specifications requirements will continue until the effluent sump pumps are repaired. (SER 92-112)

6. MWRs 914507, 915910 and Associated ICPs. Deenergize Reactor Protection and Safeguards Control Power While Shut Down. The MWRs replace the ESF test switch blocks in the "B" train of safeguards (PC-429C-XBT and PC-468A-XBT) and require deenergizing the "B" train of safeguards control power.

The safeguards and reactor protection as-built project involves hand-over-hand wire tracing in the respective protection and safeguards racks. To reduce the chance for electric shock, reactor protection control power and safeguards control power are deenergized when wire tracing is performed. Both trains of reactor protection control power are deenergized simultaneously, and one train of safeguards control power is deenergized, during performance of the wire tracing.

Summary of Safety Evaluation: Per Technical Specifications, if two trains of reactor protection are not operable, the unit must be maintained in at least hot shutdown. For the wire tracing activities, both trains of RPS are simultaneously deenergized. To ensure no positive reactivity is added from the control rods, control rod drive mechanisms are red tagged deenergized. One of the following sets of breakers is tagged open or racked out: Reactor trip and bypass breakers; or rod drive MG supply breakers; or rod drive MG load breakers.

The only Technical Specification requirement for safeguards functions during refueling is that containment ventilation isolation (CVI) be operable during refueling operations. Prior to deenergizing a train of safeguards control power, the duty shift superintendent shall determine that the opposite train automatic function is operable, and that both trains of valves operate manually from the control room, by reviewing equipment status, maintenance in progress, etc. Once this determination has been made, no maintenance to CVI components is allowed. Since service water is a shared system with the opposite unit, administrative controls are established to ensure service water system operability for the operating unit. For this work, the miscellaneous relay rack is not deenergized and continues to perform its control function in full. No other functions are a safety concern during cold shutdown or refueling operations. (SER 92-033)

7. MWRs 922392, 922393, 921803, 922394, Pressurizer Spray Valve Bellows Leak Detection Lines. The Unit 1 and 2 pressurizer spray valves bellows leak detection lines are revised by removing the 3/8" Whitey valves and the Swagelok fittings on each side of the valves. This change reduces the cantilevered distance of the line and weight which should eliminate the fatigue failure of the lines identified in condition report 92-192.

Summary of Safety Evaluation: The leak detection line change reduces the cantilevered distance of the line by approximately 6" and the weight by approximately 75%. This change also significantly reduces the stress in the line, which is expected to eliminate the vibration cracking of the lines. The pressure gauge does not require periodic calibration or isolation since the probability of a break in the modified line is considered very low and would only be of concern if the bellows were damaged, removal of the valves is acceptable.

Failure of the pressure indication line could not cause a reactor coolant leak. The line would only see reactor coolant if the spray valve bellows developed a leak. Then if the line failed, a small leak (<1/4" diameter) would be created. This size leak is well within charging pump capacity. (SER 92-047)

8. MWR 922643, Spent Fuel Pool Bridge. MWR 922643 removes an abandoned nitrogen line and valves that was used to supply nitrogen to the fuel sipper.

Summary of Safety Evaluation: The abandoned nitrogen line runs from El. 8' near the south truck access in the PAB, up to El. 66' of the spent fuel pool (SFP) west trench track. The line is adequately supported up to the SFP, but in the SFP west trench track the line is unsupported. Isolation valves NG-1663C and NG-1663D for the line are also removed. The line is removed up to valve NG-1663B where it is capped with a Swagelok fitting. A post installation leak test is performed to verify the integrity of the capped connection.

Removal does not violate QA or seismic boundaries or effect nuclear safety. Removal takes place during normal power operation and does not require any system to be out of service. (SER 92-051)

9. MWR 924734, SMP 1110. As-Built Wire Tracing of Unit 2 Turbine Generator Circuitry in Main Control Board C02. The main control board as-built project includes hand-over-hand wire tracing of the Unit 2 turbine generator, exciter and main breaker circuitry within section C02.

Wires are traced from Unit 2 turbine generator components in section C02 up to their termination points. This requires wire tracing in Risers 41 and 42. Also present in these risers is wiring for various Unit 2 alarms, 2X11 and 2X12 ammeters, 2B01 and 2B02 voltmeters, Unit 1 fault recorder, 2B52-38B (MCC 2B32 normal supply), 2B52-40C (2B03 to 2B04 tie), 2B52-44B (2B02 normal feed), 2B52-45B (2B01 normal feed), and 2A52-72 (2A05 to 2A06 tie). The other wires are traced from their riser terminals to the point where they exit the risers, where they are labeled to indicate wire number and termination point. This information is used for future wire tracing in C02 and ensures that wiring in Risers 41 and 42 is disturbed only once.

Summary of Safety Evaluation: Main control board C02 contains common electrical controls, which include: Unit 1 and 2 turbine generators; 345 kV, 13.8 kV, Unit 1 and 2 4160 V; Unit 1 and 2 480 V; emergency diesel generators; and gas turbine. Only Unit 2 turbine generator-related components were selected to be traced. This prevents other systems from being directly affected. With the exception of four 480 V breakers and one 4160 V breaker, these systems are not indirectly affected because they do not have wires in Risers 41 or 42 and are physically separated from the turbine generator related circuitry, which is confined to the north end of C02.

Affected circuitry is fully tested upon restoration per special maintenance procedures (SMP) 1112, 1113, and 1114. This ensures that proper turbine trip function and other affected circuits were restored. Also, any disturbed wire bundles are physically inspected upon completion. (SER 92-086)

10. MWR 924736. Inspection and Possible Replacement of CV-350 Discs. The MWR unbolts and separates the bonnet from the body of emergency borate valve, 2CV-350 to inspect the discs which are believed to be warped. Replacement parts are available if existing parts do not conform to standards. MWR 924736 was initiated in conjunction with MR 91-133*A during U2R18. The interim conditions during the shutdown isolates the system to allow it to be flushed and drained, and heat tracing deenergized.

Summary of Safety Evaluation: In the isolation for this maintenance, the charging system is out of service. Maintenance on the valve is accomplished in accordance with the component instruction manual. When the valve is returned to service, the replaced parts have no affect on the operation of the boric acid system. Post-maintenance testing is accomplished with the motor-operated valve (MOV) diagnostic system and does not impact operation of the system. A boration flow path from the boric acid storage tank (BAST) or refueling water storage tank (RWST) to the reactor coolant system through the safety injection pumps or the refueling water storage tank (RWST) to the reactor coolant system through the residual heat removal pumps exist at all times in accordance with TS requirements. (SER 92-092)

12. U2C19 Reload Core. Point Beach Unit 2 Cycle 19 Fuel Reload. The Unit 2 Cycle 19 reload contains 12 fresh Region 21A upgraded optimized fuel assemblies (OFA) at 3.8 w/o, 16 fresh Region 21B upgraded OFA at 4.2 w/o, 16 Region 20A upgraded OFA, 12 Region 20B upgraded OFA, 12 Region 19A upgraded OFA, 16 Region 19B upgraded OFA, 12 Region 18A upgraded OFA, 16 Region 18B upgraded OFA, 8 Region 17B OFA, and 1 Region 11A standard fuel assembly. The Cycle 19 core is the fourth reload containing a full region of upgraded OFA fuel for Unit 2.

Summary of Safety Evaluation: The core design was performed assuming the reactor coolant system can be operated at a pressure of either 2000 or 2250 psia. As a result of the Cycle 19 evaluation, it is concluded that the Cycle 19 design does not cause previously acceptable safety limits to be exceeded, provided that: a) Cycle 18 burnup is bounded by 10250 and 11250 MWD/MTU; b) Cycle 19 burnup is limited to the end-of-full-power-capability (EOFPC, which is defined as the burnup of fuel when all control rods are fully withdrawn, and less than or equal to 10 ppm of boric acid at the Cycle 19 rated power condition of 1518.5 MWt) plus 1500 MWD/MTU power coastdown operation; and c) There is adherence to the plant operating limitations given in the Technical Specifications. (SER 92-088)

13. U1C20 Reload Core. The Unit 1 Cycle 20 reload contains 12 fresh Region 22A upgraded optimized fuel assemblies (OFA) at 3.8 w/o, 16 fresh Region 22B upgraded OFA at 4.0 w/o, 12 Region 21A upgraded OFA, 16 Region 21B upgraded OFA, 16 Region 20A upgraded OFA, 12 Region 20B upgraded OFA, 1 Region 20C upgraded OFA, 12 Region 19A upgraded OFA, 15 Region 19B OFA, 4 Region 18A OFA, 4 Region 18B OFA, and 1 Region 8 standard fuel assembly.

Summary of Safety Evaluation: The core design was performed assuming the reactor coolant system can be operated at a pressure of either 2000 or 2250 psia. As a result of the Cycle 20 evaluation, it is concluded that the Cycle 20 design does not cause previously acceptable safety limits to be exceeded, provided that: a.) Cycle 19 burnup is bounded by 10600 and 11600 MWD/MTU; b.) Cycle 20 burnup is limited to the end-of-full-power-capability (EOFPC, which is defined as the burnup of fuel when all control rods are fully withdrawn, and less than or equal to 10 ppm of residual boron at the Cycle 20 rated power condition of 1518.5 MWt) plus 1500 MWD/MTU power coastdown; c.) There is adherence to the plant operating limitations given in the Technical Specifications; d.) The safety aspects on the reactor internals of utilizing PPSAs have been assumed by WE; and e.) The effect of the Cycle 20 design of the Boron Dilution Event in Cold Shutdown has been assumed by Wisconsin Electric Power. (SER 92-038)

U1C20 Reload Core Revision 1.

Summary of Safety Evaluation: Fuel assembly H32 was found to be defective and was replaced with assembly N08. Fuel assembly V17 was found to have a damaged fuel rod and was reconstituted (the damaged rod was replaced with a SS rod).

Fuel assembly H32 was not used as previously planned because of some elongation and slight discoloration of one of the fuel rods. Assembly N08 was used in place of H32. It was also discovered that a fuel rod in assembly V17 was damaged. Assembly V17 was reconstituted by removing the top nozzle on the assembly, replacing the damaged fuel rod with a stainless steel rod and putting the top nozzle back on the assembly.

Evaluations of the nuclear and thermal hydraulic performance of the U1C20 core were performed by the NSSS supplier for these changes. The evaluation verified that the changes do not affect the conclusions of the previous safety evaluation and the DNB design basis is not changed. The non-LOCA analyses remain valid and bounding for the U1C20 core with these changes.

Based on the above, these changes do not affect the conclusions of the previous safety evaluations, the conclusions of the initial safety evaluation remain valid.
(SER 92-038-01)

V. NUMBER OF PERSONNEL AND PERSON-REM BY WORK GROUP AND JOB FUNCTION - 1992

Job Group Station Employees	Number of Personnel Greater Than 100 mrem	Total rem for Job Group	Work Function and Total Person-rem					
			Reactor Operations & Surveillance	Routine Maintenance	Inspections	Special Maintenance	Waste Processing	Refueling
Operations	73	28.450	18.400	-------	7.110	-------	0.200	2.740
Maintenance	43	56.640	-------	38.020	2.390	-------	-------	16.230
Chemistry & Health Physics	26	19.390	18.540	-------	-------	-------	0.850	-------
Instrumentation & Control	15	3.350	-------	2.030	0.340	0.080	-------	0.900
Technical Services	2	0.530	0.150	-------	0.100	-------	-------	0.280
Administration & Engineering, Regulatory Services	9	2.590	0.090	-------	2.500	-------	-------	-------
Utility Employees	28	28.100	1.210	25.710	1.180	-------	-------	-------
Contractor Workers & Others	231	117.370	0.150	-------	30.040	86.040	1.140	-------
GRAND TOTALS	427	256.420	38.540	65.760	43.660	86.120	2.190	20.150

POINT BEACH NUCLEAR PLANT CALENDAR YEAR 1992		
Whole Body Exposure Range (rem)	Total Number of Individuals	Total Exposure Received (rem)
No Measurable Exposure	460	0.000
Less than 0.100	190	7.420
0.100 to 0.250	119	19.860
0.250 to 0.500	124	40.840
0.500 to 0.750	69	41.850
0.750 to 1.000	41	34.600
1.000 to 2.000	70	103.100
2.000 to 3.000	4	8.750
3.000 to 4.000	0	0.000
4.000 +	0	0.000
GRAND TOTALS	1077	256.420

766 individuals were monitored exempt from the provisions of 10 CFR 20.
This report meets the requirements of 10 CFR 20.407(a)(1).

VI. STEAM GENERATOR EDDY CURRENT TESTING

UNIT 1

Inspection Plan: During the Unit 1 Refueling 19 outage, eddy current testing was performed from April 15, 1992, to April 18, 1992. An approximate 20% sample was inspected full length and an additional first support sample was done around the peripheral to address loose parts concerns. The extent tested in each steam generator is as follows:

Eddy Current Inspection Plan		
Extent of Inspection	Number of Tubes	
	"A" SG	"B" SG
Full Length	591	592
No. 6 TSP	54	54
No. 1 TSP	62	90
Totals	707	736

Inspection Results: The results of these inspections revealed 12 tubes in the "A" steam generator with reportable indications, and 5 in the "B" steam generator. The following is a summary of the eddy current inspection results listing the largest reportable indication per tube:

Eddy Current Inspection Results Hot Leg (Cold Leg)		
	"A" SG	"B" SG
20-29%	1	1
MBM	5 (2)	3 (1)
MMB	43 (1)	0
NTE	1	0
Totals	10 (3)	4 (1)

% - Percent Through Wall Indication
MBM - Manufacturing Burnishing Marks
MMB - Multiple MBMs
NTE - No Tube Expansion

Repaired or Plugged Tubes: Steam generator tube plugging was not performed on either steam generator this outage as a result of eddy current indications.

Tubes with Indications - Not Plugged: The following is a list of tubes which had indications but were not repaired or plugged as a result of eddy current testing during U1R19.

"A" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
PBH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
6-25	MMB	3H	0.1"
6-25	MMB	2H	3.7"
10-12	BLG	TSH	0.2"
14-24	MMB	1C	13.4"
14-24	MMB	4C	19.5"
14-24	MMB	5H	37.8"
16-22	MMB	4H	37.8"
19-54	19	AV4	0.0"
22-60	MBM	BPC	4.4"
24-26	MBM	1H	47.3"
24-26	MBM	1H	45.3"
25-84	MBM	4H	26.1"
28-11	14	AV3	0.0"
31-13	13	AV3	0.0"
32-14	12	AV3	0.0"
34-62	MBM	1C	44.1"
34-63	MBM	4C	35.3"
35-18	11	AV3	0.0"
35-18	13	AV2	0.0"
37-20	14	AV2	0.0"
37-21	15	AV2	0.0"
38-22	11	AV2	0.0"
38-22	12	AV3	0.0"
38-69	NTE	TSH	0.0"

Row - Column	Indication	Location	Inch Mark
41-29	10	AV3	0.0"
41-29	13	AV2	0.0"
41-48	MBM	2H	49.5"
42-32	12	AV4	0.0"
42-32	13	AV3	0.0"
42-32	12	AV2	0.0"
42-32	MBM	3H	41.5"
43-59	MBM	3H	21.5"
45-43	28	AV1	0.0"

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
PBH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

Row - Column	Indication	Location	Inch Mark
9-34	MBM	3C	46.1"
21-42	MBM	TSH	43.5"
24-42	MBM	4H	5.3"
26-84	15	AV3	0.0"
35-51	15	AV1	0.0"
35-51	15	AV3	0.0"
35-51	27	AV2	0.0"
35-75	14	AV3	0.0"
36-74	14	AV3	0.0"
36-74	18	AV2	0.0"
37-73	16	AV3	0.0"
43-60	BLG	TSH	0.2"
44-49	BLG	TSH	0.2"
45-41	MBM	5H	25.5"
45-41	MBM	5H	26.3"

UNIT 2

Inspection Plan: During the Unit 2 Refueling 18 outage, eddy current testing was performed October 9, 1992, to October 21, 1992. An approximate 20% sample (including sleeved and unsleeved tubes) was full length inspected. The unsleeved hot leg tubes were inspected and peripheral tubes in both steam generators were examined to the first support plate to address loose parts concerns. As a result of findings in the "B" steam generator, an additional 2S (40%) sample was performed. The extent tested in each steam generator is as follows:

Eddy Current Inspection Plan		
Extent of Inspection	Number of Tubes	
	"A" SG	"B" SG
Full Length	253	281
No. 1 TSP	1465	3034
No. 6 TSP	76	1278
SLEEVES	600	560
RPC	78	86
Totals	2472	5239

RPC - Rotating write-up pancake coil inspections done to the extent necessary to bound distorted indications.

Inspection Results: The results of these inspections revealed 53 tubes in the "A" steam generator with reportable indications, and 131 in the "B" steam generator. The following is a summary of the eddy current inspection results listing the largest indication per tube:

Eddy Current Inspection Results Hot Leg (Cold Leg)		
	"A" SG	"B" SG
DI	(6)	
20-29%	4 1(9)	(44)
30-39%	1 (3)	7 (21)
40-49%		(1)
≥50%		2
Axial Ind.	36	45
MBM	1	
Totals	42 (18)	65 (66)

% - Percent Through Wall Indication
DI - Distorted Indication
MBM - Manufacturing Burnishing Marks

Repaired or Plugged Tubes: The following lists the tubes which were mechanically plugged during the Unit 2 Refueling 18 outage:

NOTE: All inch marks are above the referenced location unless otherwise specified.

Plugged Tube in the "A" Steam Generator		
Row - Column	Indication/ %	Location
3-2	MAI	2.1" TRN
2-7	MAI	3.0" TEH
8-9	SAI	7.2" TEH
2-10	MAI	5.1" TEH
10-11	MAI	8.3" TEH
10-12	SAI	9.8" TEH
5-12	SAI	6.9" TEH
21-13	SAI	8.1" TEH
19-13	SAI	7.9" TEH
3-13	SAI	8.3" TEH
13-15	SAI	6.8" TEH
33-19	MAI	6.7" TEH
35-28	MAI	0.1" TRH
34-36	MAI	0.0" TRH
40-38	SAI	0.1" TRH
40-38	MAI	6.0" TRH
42-47	MAI	4.1" TEH
44-47	SAI	3.5" TEH
33-55	MAI	3.3" TEH
36-56	SAI	5.1" TEH
34-58	MAI	5.8" TEH
34-64	SAI	3.2" TEH
32-64	SAI	2.0" TEH
38-65	MAI	2.2" TEH
37-65	SAI	6.4" TEH
36-65	MAI	3.0" TEH
40-66	SAI	4.6" TEH
34-66	MAI	7.2" TEH

Plugged Tube in the "A" Steam Generator		
Row - Column	Indication/ %	Location
38-67	MAI	0.6" TEH
34-70	SAI	6.4" TEH
33-73	MAI	4.4" TEH
6-74	SAI	16.6" TEH
26-77	SAI	9.0" TEH
22-77	SAI	7.6" TEH
19-80	SAI	3.2" TEH
13-81	MAI	5.5" TEH
7-88	SAI	9.1" TEH

% - Percent Through Wall
1H or 1C - Support Plate Number Hot or Cold Leg
MAI - Multiple Axial Indication
SAI - Single Axial Indication
TEH - Tube End Hot Leg
TSH - Tube Sheet Hot Leg

Plugged Tube in the "B" Steam Generator		
Row - Column	Indication/ %	Location
15-5	SAI	10.6" TEH
1-5	MAI	4.7" TEH
19-9	MAI	6.6" TEH
17-9	MAI	5.6" TEH
16-9	SAI	3.0" TEH
11-9	SAI	12.9" TEH
15-11	SAI	2.4" TEH
6-12	MAI	5.8" TEH
29-14	73%	0.0" TEH
26-15	MAI	2.3" TEH
23-15	MAI	12.6" TEH
20-15	SAI	3.3" TEH
24-21	69%	33.5" TSH
37-28	22%	0.0" 1C

Plugged Tube in the "B" Steam Generator		
Row - Column	Indication/ %	Location
39-33	SAI	14.7" TEH
43-34		0.0"
44-35		0.0"
41-35	SAI	7.2" TEH
44-36		0.0"
43-36		0.0"
37-41	16%	0.1" 1C
38-42	MAI	6.2" TEH
1-42	MAI	4.0" TEH
42-43	MAI	5.4" TEH
1-47	SAI	3.5" TEH
32-48	42%	0.0" 1C
1-51	MAI	3.3" TEH
1-52	SAI	3.4" TEH
1-59	SAI	3.6" TEH
37-60	51%	6.7" TEH
39-61	20%	0.0" 1C
33-63	SAI	6.8" TEH
30-63	MAI	8.0" TEH
37-64	SAI	3.1" TEH
33-66	MAI	11.4" TEH
36-68	MAI	9.4" TEH
27-68	MAI	8.5" TEH
37-70	MAI	7.1" TEH
33-71	15%	0.0" 1C
1-72	SAI	4.1" TEH
23-73	MAI	7.2" TEH
5-74	SAI	15.0" TEH
33-74	MAI	7.6" TEH
23-76	MAI	2.2" TEH
26-77	SAI	9.4" TEH

Plugged Tube in the "B" Steam Generator		
Row - Column	Indication/ %	Location
22-77	SAI	8.4" TEH
13-80	SAI	5.5" TEH
24-80	MAI	5.6" TEH
22-81	SAI	1.6" TEH
2-84	RST	0.0" 6H
22-85	25%	2.7" TEH
4-85	SAI	3.9" TEH
19-86	MAI	2.0" TEH

% - Percent Through Wall
1H or 1C - Support Plate Number Hot or Cold Leg
MAI - Multiple Axial Indication
SAI - Single Axial Indication
TEH - Tube End Hot Leg
TSH - Tube Sheet Hot Leg
RST - Restricted

Tubes with Indications - Not Plugged: The following is a list of tubes with indications not repaired during Unit 2 Refueling 18 outage as a result of eddy current indications.

"A" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
1-26	DIN	TEH	4.0"
1-91	28	1C	0.0"
2-7	DI	TEH	2.7"
2-7	MAI	TEH	3.0"
2-7	MAI	TEH	3.0"
2-10	MAI	TEH	5.1"
2-10	MAI	TEH	5.1"

Row - Column	Indication	Location	Inch Mark
2-10	MAI	TEH	5.1"
2-10	MAI	TEH	5.1"
3-2	DRI	TEH	2.1"
3-2	MAI	TRN	2.1"
3-2	MAI	TRN	2.1"
3-2	MAI	TRN	2.1"
3-2	MAI	TRN	2.1"
3-13	DI	TEH	7.9"
3-13	SAI	TEH	8.3"
3-13	SAI	TEH	8.3"
3-85	DIN	TEH	13.6"
3-92	30	TSC	20.1"
4-31	18	TSC	1.0"
4-34	16	TSC	2.0"
4-34	21	TSC	1.6"
5-12	SAI	TEH	6.9"
5-12	SAI	TEH	6.9"
5-12	DI	TEH	6.5"
5-35	14	TSC	1.2"
5-88	DIN	TEH	12.6"
6-25	21	TSC	0.7"
6-33	24	TSC	0.5"
6-35	12	TSC	0.7"
6-74	DI	TEH	14.4"
6-74	SAI	TEH	16.6"
6-74	SAI	TEH	16.6"
6-74	DI	TEH	17.0"
6-74	SAI	TEH	13.9"
6-74	SAI	TEH	13.9"
7-1	30	1C	0.0"
7-34	21	TSC	0.5"
7-35	14	TSC	0.4"

Row - Column	Indication	Location	Inch Mark
7-35	16	TSC	0.9"
7-35	14	TSC	1.6"
7-37	27	TSC	0.5"
7-78	19	TSH	0.6"
7-88	SAI	TEH	9.1"
7-88	SAI	TEH	9.1"
7-88	SAI	TEH	4.2"
7-88	SAI	TEH	4.2"
8-6	DIN	TEH	15.7"
8-9	SAI	TEH	7.2"
8-9	SAI	TEH	7.2"
8-9	DI	TEH	8.9"
8-34	DI	TSC	0.5"
8-34	10	TSC	2.0"
8-36	DI	TSC	0.5"
8-37	DI	TSC	0.8"
9-28	37	TSC	0.4"
9-35	12	TSC	0.5"
9-37	DI	TSC	0.4"
9-38	16	TSC	1.8"
9-47	22	TSC	1.3"
9-48	DI	TSC	0.8"
10-11	MAI	TEH	8.3"
10-12	DI	TEH	10.4"
10-12	SAI	TEH	9.8"
10-28	21	TSC	0.7"
10-66	17	TSC	0.5"
10-86	DIN	TEH	6.4"
11-47	13	TSC	0.7"
12-89	25	1H	0.0"
13-15	SAI	TEH	6.8"
13-15	DI	TEH	7.0"

Row - Column	Indication	Location	Inch Mark
13-47	20	TSC	0.9"
13-81	MAI	TEH	5.5"
13-81	DI	TEH	9.7"
13-81	SAI	TEH	1.7"
16-30	19	TSC	0.6"
17-47	18	TSC	0.6"
18-5	21	1H	0.0"
18-6	35	1H	0.0"
18-17	DIN	TEH	6.0"
18-47	10	TSC	0.8"
18-60	14	TSC	11.8"
19-13	SAI	TEH	7.9"
19-13	DI	TEH	6.2"
19-80	SAI	TEH	3.2"
20-37	DI	TSC	0.7"
20-53	12	TSC	2.0"
21-13	SAI	TEH	8.1"
21-13	DI	TEH	7.3"
22-7	27	2H	0.0"
22-77	SAI	TEH	7.6"
23-73	DIN	TEH	8.1"
23-76	MBM	TEH	8.9"
23-76	MBM	TEH	12.0"
24-72	DIN	TEH	19.9"
26-77	SAI	TEH	9.0"
27-15	DIN	TEH	11.6"
31-16	DIN	TEH	9.0"
31-69	DIN	TEH	8.2"
32-16	DIN	TEH	7.4"
32-64	SAI	TEH	2.0"
32-64	DI	TEH	4.0
33-19	DI	TEH	10.3"

Row - Column	Indication	Location	Inch Mark
33-19	MAI	TEH	6.7"
33-19	MAI	TEH	6.7"
33-55	MAI	TEH	3.3"
33-55	DI	TEH	6.0"
33-70	DRN	TEH	2.3"
33-73	MAI	TEH	4.4"
33-73	MAI	TEH	4.5"
33-73	DI	TEH	4.8"
33-73	MAI	TEH	4.4"
34-33	DIN	TEH	9.2"
34-36	MAI	TRN	0.0"
34-36	MAI	TRN	0.0"
34-36	MAI	TRN	0.0"
34-36	DRI	TEH	2.8"
34-45	DIN	TEH	18.9"
34-58	MAI	TEH	5.8"
34-58	DI	TEH	3.0"
34-64	SAI	TEH	3.2"
34-64	DI	TEH	3.3"
34-66	MAI	TEH	7.2"
34-66	MAI	TEH	7.2"
34-66	DI	TEH	7.1"
34-66	MAI	TEH	7.2"
34-70	DI	TEH	6.4"
34-70	SAI	TEH	6.4"
34-75	DIN	TEH	18.4"
35-28	DRI	TEH	2.8"
35-28	MAI	TRN	0.1"
36-56	SAI	TEH	5.1"
36-56	DI	TEH	6.0"
36-54	DRN	TEH	2.2"
36-65	MAI	TEH	3.0"

Row - Column	Indication	Location	Inch Mark
36-65	MAI	TEH	3.0"
37-65	SAI	TEH	6.4"
38-65	MAI	TEH	2.2"
38-65	DI	TEH	6.2"
38-65	DI	TEH	4.9"
38-65	MAI	TEH	2.2"
38-65	MAI	TEH	2.2"
38-67	MAI	TEH	0.6"
38-67	MAI	TEH	0.6"
38-67	MAI	TEH	0.6"
40-26	18	TSH	3.6"
40-26	14	TSH	4.0"
40-26	28	TSH	8.3"
40-26	12	TSH	5.3"
40-26	15	TSH	5.7"
40-26	28	TSH	9.1"
40-38	MAI	TRN	6.0"
40-38	SAI	TRN	0.7"
40-38	DRI	TRN	0.0"
40-38	SAI	TRN	0.1"
40-38	MAI	TRN	6.0"
40-66	SAI	TEH	4.6"
40-66	DI	TEH	4.8"
42-47	DI	TEH	5.5"
42-47	MAI	TEH	4.1"
42-47	MAI	TEH	4.1"
44-47	SAI	TEH	3.5"
44-47	DI	TEH	4.1"

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
1-2	25	1C	0.0"
1-5	SAI	TEH	6.2"
1-5	MAI	TEH	4.7"
1-5	MAI	TEH	4.7"
1-5	MAI	TEH	3.9"
1-5	MAI	TEH	3.9"
1-5	SAI	TEH	7.1"
1-21	17	1C	0.0"
1-42	SAI	TEH	3.2"
1-42	MAI	TEH	4.0"
1-42	MAI	TEH	4.0"
1-47	SAI	TEH	3.5"
1-50	MAI	TEH	3.3"
1-51	MAI	TEH	6.1"
1-51	MAI	TEH	3.3"
1-51	MAI	TEH	6.1"
1-52	SAI	TEH	3.4"
1-59	SAI	TEH	3.6"
1-68	DI	TEH	2.5"
1-69	DI	TEH	2.0"
1-72	SAI	TEH	4.1"
2-1	15	TSC	8.8"
2-17	17	1H	26.8"
3-1	29	1C	0.0"
3-35	DIN	TSC	1.7"
3-55	10	TSC	0.7"
4-71	20	TSC	0.4"

Row - Column	Indication	Location	Inch Mark
4-71	23	TSC	0.4"
4-85	SAI	TEH	3.9"
5-2	24	1C	0.0"
5-43	DIN	TSC	0.6"
5-74	SAI	TEH	8.4"
5-74	SAI	TEH	15.0"
6-12	MAI	TEH	5.8"
6-12	SQR	TEH	3.6"
6-15	28	TSH	39.4"
7-15	20	1H	0.0"
7-78	DI	TSH	0.4"
9-74	38	TSC	1.1"
9-81	30	1H	19.0"
11-9	DI	TEH	12.9"
11-9	SAI	TEH	12.9"
11-88	29	1C	0.0"
12-72	11	TSC	1.2"
13-4	32	1C	0.0"
13-80	SAI	TEH	7.7"
13-80	SAI	TEH	5.5"
14-37	16	TSC	0.9"
15-5	SAI	TEH	10.6"
15-11	DI	TEH	3.5"
15-11	SAI	TEH	2.4"
15-33	16	TSC	1.3"
15-33	16	TSC	1.3"
15-62	DIN	TSC	1.1"
15-62	DIN	TSC	1.1"
16-9	SAI	TEH	3.0"
16-9	DI	TEH	4.1"
17-9	MAI	TEH	5.6"
17-9	DI	TEH	5.9"

Row - Column	Indication	Location	Inch Mark
17-9	MAI	TEH	5.6"
17-9	MAI	TEH	5.6"
17-9	SAI	TEH	7.7"
17-9	SAI	TEH	10.2"
17-21	36	6H	0.0"
17-22	13	1C	0.2"
17-22	13	1C	0.2"
18-87	14	3H	0.0"
19-9	MAI	TEH	6.6"
19-9	DI	TEH	6.7"
19-31	31	TSC	0.4"
19-31	31	TSC	0.4"
19-37	22	TSC	0.3"
19-37	22	TSC	0.8"
19-86	MAI	TEH	2.0"
19-86	MAI	TEH	2.1"
19-86	MAI	TEH	2.0"
20-15	SAI	TEH	3.3"
20-57	DIN	TSC	0.8"
20-57	DIN	TSC	0.8"
20-58	20	TSC	1.0"
20-58	20	TSC	1.0"
20-68	14	TSC	0.8"
20-68	14	TSC	0.8"
20-69	DIN	TSC	0.4"
21-7	24	1C	0.0"
21-31	15	TSC	1.0"
21-31	15	TSC	1.0"
22-19	22	TSH	30.4"
22-24	23	1C	0.0"
22-28	30	1C	0.0"
22-77	SAI	TEH	8.4"

Row - Column	Indication	Location	Inch Mark
22-81	SAI	TEH	1.6"
22-85	SAI	TEH	2.1"
22-85	25	TEH	2.7"
22-86	33	1C	0.0"
23-15	SAI	TEH	4.0"
23-15	SAI	TEH	6.5"
23-15	MAI	TEH	12.6"
23-29	31	1C	0.0"
23-29	31	1C	0.0"
23-69	35	1C	0.0"
23-73	MAI	TEH	7.2"
23-76	DI	TEH	8.5"
23-76	MAI	TEH	2.2"
23-85	DI	3C	0.0"
24-21	69	TSH	33.5"
24-32	13	TSC	2.5"
24-32	13	TSC	2.5"
24-44	17	TSC	0.6"
24-80	MAI	TEH	8.0"
24-80	MAI	TEH	5.6"
24-85	DI	6H	0.0"
26-12	12	1C	0.0"
26-13	31	1C	0.0"
26-15	DI	TEH	2.7"
26-15	MAI	TEH	2.3"
26-24	23	1C	0.0"
26-51	37	1C	0.0"
26-62	10	TSC	5.8"
26-62	10	TSC	5.8"
26-63	39	AV3	0.0"
26-63	39	AV3	0.0"
26-75	33	1H	14.3"

Row - Column	Indication	Location	Inch Mark
26-77	SAI	TEH	10.1"
26-77	SAI	TEH	3.7"
26-77	SAI	TEH	7.0"
26-77	SAI	TEH	9.4"
27-24	28	AV2	0.0"
27-29	20	1C	0.0"
27-29	20	1C	0.0"
27-51	DIN	1C	0.0"
27-68	MAI	TEH	8.5"
27-68	SAI	TEH	5.6"
27-69	18	1C	0.0"
27-79	19	TSH	6.3"
27-79	28	TSH	3.8"
28-16	24	TSH	47.8"
28-16	32	TSH	43.8"
28-23	DI	TSH	0.7"
28-35	24	4H	0.0"
28-35	24	4H	0.0"
28-39	22	1C	0.0"
28-51	DIN	1C	0.0"
29-14	73	6H	0.0"
29-16	38	1C	0.0"
29-19	35	1C	0.0"
29-26	27	1C	0.0"
29-28	17	1C	0.0"
29-30	12	1C	0.0"
30-35	33	1C	0.0"
30-35	33	1C	0.0"
30-47	22	1C	-0.1"
30-48	24	1C	0.0"
30-49	26	1C	0.0"
30-50	20	1C	0.0"

Row - Column	Indication	Location	Inch Mark
30-63	MAI	TEH	8.0"
31-28	16	1C	0.0"
31-47	DIN	1C	0.0"
31-66	35	1C	0.0"
31-67	14	1C	0.0"
32-20	20	1C	0.0"
32-21	16	1C	0.0"
32-48	42	1C	0.0"
32-53	DIN	1C	0.0"
32-60	38	1C	-0.1"
32-63	DIN	1C	0.0"
32-70	27	1C	0.0"
32-74	DI	1C	0.0"
33-19	17	1C	0.0"
33-37	DI	TSH	0.3"
33-46	26	1C	0.0"
33-48	DIN	1C	0.1"
33-48	DI	TSH	0.6"
33-51	10	1C	0.0"
33-58	21	1C	0.0"
33-60	27	1C	0.0"
33-63	SAI	TEH	6.8"
33-63	DI	TEH	6.6"
33-66	MAI	TEH	11.4"
33-66	SAI	TEH	12.1"
33-71	SAI	TEH	6.7"
33-71	15	1C	0.0"
33-73	21	1C	0.0"
33-74	DI	1C	0.0"
33-74	MAI	TEH	7.6"
34-23	24	1C	0.0"
34-25	DIN	1C	0.0"

Row - Column	Indication	Location	Inch Mark
34-39	34	1C	0.0"
34-52	32	1C	0.0"
34-52	22	1C	0.0"
34-53	27	1C	0.0"
34-55	32	1C	-0.0"
34-67	16	1C	0.0"
35-58	DI	TSH	4.8"
35-58	DI	TSH	36.2"
36-19	14	1C	0.0"
36-21	25	1C	0.0"
36-22	23	1C	0.0"
36-25	23	1C	0.0"
36-30	20	TSH	3.0"
36-48	16	TSH	41.6"
36-48	13	TSH	34.6"
36-63	12	1C	0.0"
36-65	10	1C	0.0"
36-66	30	1C	0.0"
36-68	SAI	TEH	14.3"
36-68	MAI	TEH	9.4"
37-21	15	1C	0.0"
37-23	29	1C	0.0"
37-25	DIN	1C	0.0"
37-28	MAI	TEH	7.4"
37-28	22	1C	0.0"
37-41	SAI	6H	0.0"
37-41	47	6H	0.0"
37-41	16	1C	0.1"
37-50	20	1C	0.0"
37-60	SAI	TEH	6.9"
37-60	51	TEH	6.7"
37-61	27	1C	0.0"

Row - Column	Indication	Location	Inch Mark
37-62	26	1C	0.0"
37-63	27	1C	0.0"
37-64	SAI	TEH	3.1"
37-67	14	1C	0.0"
37-68	25	1C	0.0"
37-70	MAI	TEH	8.3"
37-70	MAI	TEH	7.1"
38-25	34	TSH	50.6"
38-42	SAI	TEH	10.1"
38-42	MAI	TEH	6.2"
38-48	30	1C	0.0"
34-49	22	1C	0.0"
38-52	39	1C	-0.1"
38-52	36	1C	0.0"
38-54	26	1C	0.0"
38-55	13	1C	-0.0"
38-61	28	AV4	0.0"
39-24	20	1C	0.0"
39-25	25	1C	0.0"
39-33	SAI	TEH	14.7"
39-34	29	1C	0.0"
39-35	34	1C	0.0"
39-61	20	1C	0.0"
39-61	SAI	TEH	8.5"
39-61	DI	TEH	8.4"
39-64	24	1C	0.0"
39-65	16	1C	0.0"
40-57	16	1C	0.0"
40-64	23	TSH	50.8"
41-35	SAI	TEH	7.2"
41-46	26	1C	0.0"
41-48	26	1C	0.0"

Row - Column	Indication	Location	Inch Mark
42-43	SAI	TEH	5.0"
42-43	MAI	TEH	5.4"
42-43	SAI	TEH	10.0"
42-49	18	1C	0.0"
42-60	18	TSH	0.9"
42-60	16	TSH	5.9"
43-58	27	TSH	2.4"
45-47	36	2H	0.0"

VII. REACTOR COOLANT SYSTEM RELIEF VALVE CHALLENGES

Overpressure Protection During Normal Pressure and Temperature Operation

There were no challenges to the Unit 1 or Unit 2 reactor coolant system power-operated relief valve or safety valves at normal operating pressure and temperature in 1992.

Overpressure Protection During Low Pressure and Temperature Operation

There were no challenges to Unit 1 or 2 power-operated relief valve during low temperature and low pressure operation in 1992.

VIII. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications during operation of Unit 1 or Unit 2 in 1992 where reactor coolant activity exceeded that allowed by Technical Specifications.