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U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
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Gentlemen:

DOCKETS 50-266 AND 50-301
ANNUAL RESULTS AND DATA REPORT - 1991
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is the Annual Results and Data Report for Point Beach Nuclear Plant, Units 1 and 2, for the year 1991. This report is submitted in accordance with Technical Specification 15.6.9.1.B pursuant to the requirements of 10 CFR 50.59(b). The report contains information regarding operational highlights of the Point Beach Nuclear Plant operations during 1991 and includes description of facility changes, tests and experiments, personnel occupational exposures, results of steam generator in-service inspections, and listings of reactor coolant system relief valve challenges.

Ten bound copies of this report are also being provided to you under separate [illegible].

Sincerely,

James Zach

James J. Zach
Vice President
Nuclear Power

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

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POWER COMPANY

POINT BEACH NUCLEAR PLANT

UNIT NOS. 1 AND 2

ANNUAL RESULTS AND
DATA REPORT
1991

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

PREFACE

This Annual Results & Data Report for 1991 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, 1991, through December 31, 1991, included a 45-day refueling/maintenance outage. Major work items included eddy current inspection of steam generators, cleaning and inspection of the containment service water system, and inspection and maintenance of the low pressure turbines. The unit experienced two trips caused by a deenergized reactor protection bus due to inverter failure; a 3-hour shutdown for the repair of low pressure feedwater heater 4A and testing of the main steam isolation and nonreturn valves; a 23-hour shutdown to replace N-[illegible] power range detector and inspect the main steam isolation valve operators; and a 76.5 hour shutdown to correct high conductivity levels in the steam generator caused by condenser tube leakage.

Unit 1 operated at an average capacity factor of 85.4% (MDC net) and an electrical/thermal efficiency of 32.2%. The unit and reactor availability were 86.0% and 87.0%, respectively. Unit 1 generated its 68 billionth kilowatt hour on February 26, 1991; its 69 billionth kilowatt hour on July 6, 1991; its 70 billionth kilowatt hour on September 26, 1991; and its 71 billionth kilowatt hour on December 23, 1991.

UNIT 2

Highlights for the period January 1, 1991, through December 31, 1991, included a 48-day refueling/maintenance outage, completing 313 days of uninterrupted operation with no significant power reductions. The unit experienced a 29.3 hour shutdown to repair steam leaks on the moisture separator reheater steam supply leakoff valve to the condenser and steam header nonreturn check valve. There was one unit trip, which resulted in a shutdown of 17.6 hours. A DC breaker termination became disconnected while pulling cable, deenergizing a reactor protection circuit.

Unit 2 operated at an average capacity factor of 86.8% (MDC net) and a net electrical/thermal efficiency of 32.5%. The unit and reactor availability were 86.4% and 87.3%, respectively. Unit 2 generated its 68 billionth kilowatt hour on February 26, 1991; its 69 billionth kilowatt hour on May 17, 1991; its 70 billionth kilowatt hour on August 6, 1991; and its 71 billionth kilowatt hour on December 19, 1991.

III. Amendments to Facility Operating Licenses

During 1991, there were three amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1 and three 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 127 to DPR-24, Amendment 131 to DPR-27, May 8, 1991: The amendments removed specific requirements for secondary source assemblies to be located in the core. In lieu of these requirements, a requirement was instituted that a minimum count rate be observed on source range instrumentation prior to criticality.

Amendment 128 to DPR-24, Amendment 132 to DPR-27, September 4, 1991: The amendments removed the organization charts from the Technical Specifications and implemented the appropriate administrative controls. Staffing and title changes were also incorporated in the change.

Amendment 129 to DPR-24, Amendment 133 to DPR-27, October 16, 1991: The amendments changed the required test frequency for the turbine stop and governor valves from monthly to annually.

IV. 10 CFR 50.59

PROCEDURE CHANGES

1. AOP-1B, (Major), Reactor Coolant Pump Malfunction, Revision 3, dated August 26, 1991. (Permanent)

   Summary of Safety Evaluation: The change converts the procedure from 1-column to 2-column format; directs a turbine trip and reactor trip prior to securing the RCPs to ensure the TS limits for operating with one RCP are not violated; consolidates corrective action steps to remove redundancy; add notes concerning the more severe RCP failure modes to heighten the operators awareness to the actions required; and deletes the time window of 10 minutes for restoring CC flow after which the RCP must be secured.

   The TS 15.3.1 limits for operating a single RCP is 3.5% reactor power. To ensure operation below this limit, AOP-1A was changed to direct tripping the reactor prior to securing an RCP. This also meets the TS limit of the reactor being shutdown if no RCPs are operating in the event that a malfunction affects both RCPs (i.e., No. 1 seal bypass line). TS 15.3.1 requires one RCP be running if the reactor is shut down if reactor temperature is >350°F unless certain conditions are maintained. This concern is addressed by directing the operators to EOP-0. EOP-0 directs a natural circulation cooldown in accordance with EOP-0.2 if neither RCP can be started.

   The format change to the procedure consolidated operator actions. This did not delete any actions for possible malfunctions. Rather, it stated the required operator actions more clearly.

   The RCP technical manual does not provide any time windows for operating an RCP without CC. The operator actions are not changed. However, a note giving the operators 10 minutes to restore CC flow is deleted. The operator actions are based

on temperature indications. If temperatures are not normal, the procedure directs securing the RCP. (SER 91-068)

2. AOP-10C, (Major), 4160 V Vital Switchgear Room Inaccessibility, Revision 0, dated May 22, 1991. (New Procedure)

Summary of Safety Evaluation: The evaluation compared all the steps (discussions, symptoms, immediate actions, required checklists, and subsequent actions) of the proposed procedure with action discussions in WE Appendix R submittals; upon which the NRC concluded that we conformed with Appendix R.

The procedure calls for steps to be taken which are contrary to the PBNP Technical Specifications. These actions include the isolation of 4160 V and 480 V safeguards buses from their normal and emergency sources, opening of diesel generator breakers, and shutdown of the emergency diesels. A note in the procedure states that certain steps in this procedure are contrary to the Technical Specifications, but are in accordance with the provisions of 10 CFR 50.54(x) and (y). (SER 91-047)

3. AOP-10D, (Major), Safe to Cold Shutdown in Alternate Shutdown Mode, Revision 0, dated May 22, 1991. (New Procedure)

Summary of Safety Evaluation: The evaluation compared all the steps (discussions, symptoms, immediate actions, required checklists, and subsequent actions) of the proposed procedure with action discussions in WE Appendix R submittals; upon which the NRC concluded that we conformed with Appendix R.

The procedure calls for steps to be taken which are contrary to the PBNP Technical Specifications. These actions include the isolation of 4160 V and 480 V safeguards buses from their normal and emergency sources, opening of diesel generator breakers, and shutdown of the emergency diesels. A note in the procedure states that certain steps in this procedure are contrary to the Technical Specifications, but are in accordance with the provisions of 10 CFR 50.54(x) and (y). (SER 91-048)

4. CSP-C.1, (Major), Response to Inadequate Core Cooling, Revision 5, dated March 21, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-024-04)

5. CSP-C.1, (Major), Response to Inadequate Core Cooling, Revision 6, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. The FSAR does not address adverse containment condition

setpoint usage and will not need to be changed due to switching the order of listing normal and adverse setpoints.

Another change made references the operator to EPIP 10.3 for operating instructions for the hydrogen recombiner. No required actions will be added by this change that were not required in the previous revision; therefore, no unresolved safety question is introduced. (SER 89-024-06)

6. CSP-C.2, (Major), Response to Degraded Core Cooling, Revision 5, dated March 21, 1991. (Permanent)

   Summary of Safety Evaluation: This revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-095-04)

7. CSP-C.2, (Major), Response to Degraded Core Cooling, Revision 6, dated October 25, 1991. (Permanent)

   Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 88-095-05)

8. CSP-C.3, (Major), Response to Saturated Core Cooling, Revision 3, dated March 21, 1991. (Permanent)

   Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-099-02)

9. CSP-C.3, (Major), Response to Saturated Core Cooling, Revision 4, dated October 25, 1991. (Permanent)

   Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-099-03)

10. CSP-H.1, (Major), Response to Loss of Secondary Heat Sink, Revision 7, dated March 7, 1991. (Permanent)

    Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-096-06)

11. CSP-H.1, (Major), Response to Loss of Secondary Heat Sink, Revision 8, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: Step 16 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33.

Changes to Steps 5, 8, 9, 12, 18 and a caution on Step 24 were made to maintain consistent nomenclature. Additional action steps were inserted in the "response not obtained" column for Steps 8.d.2, 8.d.3 and 8.d.5 since alternate methods of operating valves are available.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The load rating for 2P-2C was revised from 104 kW to 91 kW to reflect FSAR values. (SER 88-096-07)

12. CSP-H.2, (Major), Response to Steam Generator Overpressure, Revision 3, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-025-02)

13. CSP-H.3, (Major), Response to Steam Generator High Level, Revision 2, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The list format in Step 9 was changed to agree with the EOP Writer's Guide. (SER 89-079-01)

14. CSP-H.4, (Major), Response to Loss of Normal Steam Release Capabilities, Revision 2, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-077-01)

15. CSP-H.5, (Major), Response to Steam Generator Low Level, Revision 3, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. The FSAR does not address adverse containment condition setpoint usage and will not need to be changed due to switching the order of listing normal and adverse setpoints. (SER 89-026-02)

16. CSP-I.1, (Major), Response to High Pressurizer Level, Revision 2, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-078-01)

17. CSP-I.1, (Major), Response to High Pressurizer Level, Revision 3, dated May 24, 1991. (Permanent)

Summary of Safety Evaluation: The revision reduces the high pressurizer level reactor trip setpoint from 90% to 80% and results in entering CSP, CSP-I.1, "Response to High Pressurizer Level," from ST-6 sooner. Step 9 requires lowering pressurizer level <80% instead of 90%. These changes are more conservative for this procedure.

Technical Specification 15.2.3 requires pressurizer level reactor trip ≤95% of span. A reactor trip set at 80% complies with this specification. (SER 89-078-02)

18. CSP-I.3, (Major), Response to Voids in Reactor Vessel, Revision 4, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-097-03)

19. CSP-I.3, (Major), Response to Voids in Reactor Vessel, Revision 3, dated October 26, 1989 (performed April 15, 1991). (Temporary)

Summary of Safety Evaluation: The revision changes Figure 1 to correspond with changes to EOP Setpoint C.1. Setpoint C.1 was revised in a more conservative direction to correspond to Amendment Nos. 129 and 133 to the Technical Specifications. (SER 88-097-05)

20. CSP-I.3, (Major), Response to Voids in Reactor Vessel, Revision 4, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes Figure 1 to correspond with changes to EOP Setpoint C.1. Setpoint C.1 was revised in a more conservative direction to correspond to Amendment Nos. 129 and 133 to the Technical Specifications. (SER 88-097-04)

21. CSP-I.3, (Major), Response to Voids in Reactor Vessel, Revision 5, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

Another change references the operator to EPIP 10.3 for operating instructions for the hydrogen recombiner. Instructions for the hydrogen recombiner have recently been developed. No required actions were added in this change that were not required in the previous revision. (SER 88-097-06)

22. CSP-P.1, (Major), Response to Imminent Pressurized Thermal Shock Condition, Revision 5, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-027-05)

23. CSP-P.1, (Major), Response to Imminent Pressurized Thermal Shock Condition, Revision 6, dated May 24, 1991. (Permanent)

Summary of Safety Evaluation: Step 21 requires the operator to depressurize RCS until either RCS subcooling <45°F [90°F] based on core exit thermocouples or pressurizer level >90% [72%]. The purpose of securing depressurization as stated in the ERG background document is to maintain a steam bubble to facilitate further pressure control. The change ensures a larger steam bubble by changing 90% [72%] to 80% [70%]. Step 26 ensures pressurizer level is reduced to <90% [72%] which became <80% [70%].

Technical Specification 15.2.3 requires pressurizer level reactor trip ≤95% of span. A reactor trip set at 80% complies with this specification. (SER 89-027-06)

24. CSP-P.1, (Major), Response to Imminent Pressurized Thermal Shock Condition, Revision 7, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 11 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 11 does not change the intent of this procedure.

The Step 18 "response not obtained" column was changed to more closely reflect the ERG wording for this corresponding step. Confusion was caused when entering Step 18 with SI pumps secured. The change provides clarification and does not alter the current safety evaluation. (SER 89-027-07)

25. CSP-P.1, (Major), Response to Imminent Pressurized Thermal Shock Condition, Revision 8, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-027-08)

26. CSP-P.2, (Major), Response to Anticipated Pressurized Thermal Shock Condition, Revision 5, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-028-05)

27. CSP-S.1, (Major), Response to Nuclear Power Generator/ATWS, Revision 4, dated March 21, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-081-03)

28. CSP-S.1, (Major), Response to Nuclear Power Generator/ATWS, Revision 5, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the

first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-081-04)

29. CSP-Z.1, (Major), Response to High Containment Pressure, Revision 5, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-029-04)

30. ECA-0.0, (Major), Loss of All AC Power, Revision 8, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-091-07)

31. ECA-0.0, (Major), Loss of All AC Power, Revision 9, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: The procedure was revised to provide updated diesel loading data for charging pump 2P-2C from 104 kW to 91 kW and the instrument air compressor from 90 kW to 93 kW to reflect FSAR values.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

Steps 19 and 29 refer operators to a more detailed description of battery charger restoration but do not alter operator actions. (SER 88-091-08)

32. ECA-0.1, (Major), Loss of All AC Power Recovery Without SI Required, Revision 5, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-092-04)

33. ECA-0.1, (Major), Loss of All AC Power Recovery Without SI Required, Revision 6, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP

setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

Wording in Step 12 was changed to indicate changes on the control room label.

The "If, then" condition statement in Step 15, Response Not Obtained (RNO), was unnecessary since this condition was already met in referring to the response not obtained column. The statement was deleted. (SER 88-092-05)

34. ECA-0.2, (Major), Loss of All AC Power Recovery With SI Required, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-031-05)

35. ECA-0.2, (Major), Loss of All AC Power Recovery with SI Required, Revision 7, dated October 25, 1991. (Permanent)

Summary of Safety Evaluation: The procedure was revised to provide updated diesel loading data for charging pump 2P-2C from 104 kW to 91 kW and instrument air compressor from 90 kW to 53 kW to reflect FSAR values.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoints. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-031-06)

36. ECA-1.1, (Major), Loss of Containment Sump Recirculation, Revision 5, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-093-04)

37. ECA-1.1, (Major), Loss of Containment Sump Recirculation, Revision 6, dated May 24, 1991. (Permanent)

Summary of Safety Evaluation: Step 24 requires the operator to depressurize RCS until either RCS subcooling <45°F [90°F] based on core exit thermocouples or pressurizer level >90% [72%]. The purpose of securing depressurization as stated in the ERG background document is to maintain a steam bubble to facilitate further pressure control. The change ensures a larger steam bubble by changing 90% [72%] to 80% [70%].

Technical Specification 15.2.3 requires pressurizer level reactor trip ≤95% of span. A reactor trip set at 80% complies with this specification. (SER 88-093-05)

38. ECA-1.1, (Major), Loss of Containment Sump Recirculation, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: Step 9 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33, Revision 9. The change references OI-33.

Step 31 was returned to the ERG usage to eliminate operator confusion with the term "maximize charging." The plant-specific means entered in this step remain within the scope of the system operation and do not present any unreviewed safety questions. (SER 88-093-06)

39. ECA-1.2, (Major), LOCA Outside Containment, Revision 5, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-032-04)

40. ECA-2.1, (Major), Uncontrolled Depressurization of both Steam Generators, Revision 9, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-033-09)

41. ECA-2.1, (Major), Uncontrolled Depressurization of both Steam Generators, Revision 10, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 11 was changed to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 11 does not change the intent of this procedure. (SER 89-033-10)

42. ECA-2.1, (Major), Uncontrolled Depressurization of Both Steam Generators, Revision 11, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The diesel loading was revised to provide updated data for charging pump 2P-2C from 104 kW to 91 kW and instrument air compressor from 90 kW to 93 kW to reflect FSAR values. (SER 89-033-11)

43. ECA-3.1, (Major), SGTR with Loss of Reactor Coolant-Subcooled Recovery Desired, Revision 9, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. The word "be" was added to the caution statement after Step 36; this caution step was awkwardly worded without it. (SER 88-094-09)

44. ECA-3.1, (Major), SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired, Revision 10, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 4 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 4 does not change the intent of this procedure. (SER 88-094-10)

45. ECA-3.1, (Major), SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired, Revision 11, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The diesel loading was revised to provide updated data for charging pump 2P-2C from 104 kW to 91 kW and instrument air compressor from 90 kW to 93 kW to reflect FSAR values. (SER 88-094-11)

46. ECA-3.2, (Major), SGTR with Loss of Reactor Coolant-Saturated Recovery Desired, Revision 8, dated March 21, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-034-08)

47. ECA-3.2, (Major), SGTR with Loss of Reactor Coolant - Saturated Recovery Desired, Revision 9, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse

containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

Step 9.b setpoints for D.13 and D.14 at 20% and [40%], respectively, were corrected. This mistake was made with the original issue of ECA-3.2 with no justification for the deviation. Accordingly, the setpoints were corrected in this revision. (SER 89-034-09)

48. ECA-3.3, (Major), SGTR Without Pressurizer Pressure Control, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-035-06)

49. ECA-3.3, (Major), SGTR Without Pressurizer Pressure Control, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-035-07)

50. EOP-0, (Major), Reactor Trip or Safety Injection, Revision 9, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-084-08)

51. EOP-0, (Major), Reactor Trip or Safety Injection, Revision 10, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 40 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 40 does not change the intent of this procedure. (SER 88-084-09)

52. EOP-0, (Major), Reactor Trip or Safety Injection, Revision 11, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The procedure was updated to reflect diesel loading changes for charging pump 2P-2C from 104 kW to 91 kW and instrument air compressor from 90 kW to 93 kW per FSAR values.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 88-084-10)

53. EOP-0.1, (Major), Reactor Trip Response, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-085-05)

54. EOP-0.1, (Major), Reactor Trip Response, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: Since the AFW actuation is expected to occur on most reactor trips due to steam generator shrinkage, Step 1 was changed to reflect this. The resultant action of isolating AFW to the unaffected unit if AFW was actuated remains the same. (SER 88-085-06)

55. EOP-0.2, (Major), Natural Circulation Cooldown, Revision 8, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: EOP-0.2, Revision 8 changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-086-08)

56. EOP-0.3, (Major), Natural Circulation Cooldown with Steam Void in Vessel, Revision 7, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: EOP-0.3, Revision 7 changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-036-07)

57. EOP-0.3, (Major), Natural Circulation Cooldown with Steam Void in Vessel, Revision 8, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the

normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-036-08)

58. EOP-1, (Major), Loss of Reactor or Secondary Coolant, Revision 10, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-087-10)

59. EOP-1, (Major), Loss of Reactor or Secondary Coolant, Revision 11, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: Step 8 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33.

The procedure was also updated for diesel loading changes for charging pump 2P-2C from 104 kW to 91 kW and for the instrument air compressor from 90 kW to 93 kW to reflect FSAR values.

Step 11.6 results in stripping the X-17B boric acid heat tracing transformer only if the emergency diesel is overloaded. The procedure currently results in stripping X-17B regardless of diesel loading. Technical Specification 15.3.2 is not violated since this limitation is not required during a LOCA. SI flow is still required at the point in the procedure where X-17B is stripped. This SI flow will prevent boric acid crystallization until the SI line is flushed.

Step 12.b was modeled more closely to the ERG usage to eliminate confusion caused by the phrase "maximum charging." By using flow as necessary, maximum charging can still be performed when appropriate.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 88-087-11)

60. EOP-1.1, (Major), SI Termination, Revision 7, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-088-06)

61. EOP-1.1, (Major), SI Termination, Revision 8, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 3 was changed to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 3 does not change the intent of this procedure. (SER 88-088-09)

62. EOP-1.1, (Major), SI Termination, Revision 9, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The procedure was updated to reflect diesel loading changes for charging pump 2P-2C from 104 kW to 91 kW and for the instrument air compressor from 90 kW to 93 kW to reflect FSAR values.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 88-088-09)

63. EOP-1.2, (Major), Small Break LOCA Cooldown and Depressurization, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-037-06)

64. EOP-1.2, (Major), Small Break LOCA Cooldown and Depressurization, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

Placing the condition of steam generator levels from the note on Step 5 into Step 5 results in the same actions being performed prior to the change. (SER 89-037-07)

65. EOP-1.3, (Major), Transfer to Containment Sump Recirculation, Revision 8, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-089-08)

66. EOP-1.3, (Major), Transfer to Containment Sump Recirculation, Revision 9, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. The FSAR does not address adverse containment condition setpoint usage and will not need to be changed due to switching the order of listing normal and adverse setpoints. (SER 88-089-09)

67. EOP-1.4, (Major), Transfer to Containment Sump Recirculation, One Train Inoperable, Revision 4, dated March 21, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-100-08)

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

Summary of Safety Evaluation: Step 14 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33.

A change was made which shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-100-09)

69. EOP-2, (Major), Faulted Steam Generator Isolation, Revision 5, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-038-04)

70. EOP-3, (Major), Steam Generator Tube Rupture, Revision 10, dated March 7, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 88-090-09)

71. EOP-3, (Major), Steam Generator Tube Rupture, Revision 11, dated July 12, 1991. (Permanent)

Summary of Safety Evaluation: Step 10 was revised to provide the proper method of restoring a battery charger to service following safety injection initiation and/or loss of normal safeguards power per Section B of OI-33. The change references OI-33. The change to Step 10 does not change the intent of this procedure. (SER 88-090-10)

72. EOP-3, (Major), Steam Generator Tube Rupture, Revision 12, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The administrative change to Step 12.b provides uniformity throughout the EOPs when writing valve lists.

The load rating for charging pump 2P-2C and the instrument air compressor were updated to reflect FSAR values. (SER 88-090-11)

73. EOP-3.1, (Major), Post-Steam Generator Tube Rupture Cooldown Using Feedwater, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-039-06)

74. EOP-3.1, (Major), Post-Steam Generator Tube Rupture Cooldown Using Feedwater, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-039-07)

75. EOP-3.2, (Major), Post-Steam Generator Tube Rupture Cooldown Using Blowdown, Revision 6, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-085-06)

76. EOP-3.2, (Major), Post-Steam Generator Tube Rupture Cooldown Using Blowdown, Revision 7, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used.

The list format in Step 13a was changed to agree with the EOP Writer's Guide. (SER 89-085-07)

77. EOP-3.3, (Major), Post-Steam Generator Tube Rupture Cooldown Using Steam Dump, Revision 8, dated April 19, 1991. (Permanent)

Summary of Safety Evaluation: The revision changes all valve and system designators to match control board nameplates. The change satisfies the requirements of control room design review HED #439. (SER 89-040-08)

78. EOP-3.3, (Major), Post-Steam Generator Tube Rupture Cooldown Using Steam Dump, Revision 9, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: The change shuts the CST suction line manual isolation valve when the auxiliary feedwater pump supply is switched to service water and the leak rate of the check valves are no longer a concern.

An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-040-09)

79. Holtec International, HPP-10780-01, (Minor), Blackness Test Procedure, Revision 0, dated July 29, 1991. (New Procedure)

Summary of Safety Evaluation: Blackness (neutron attenuation) testing of the boraflex panels in the SFP is performed with a Californium-252 neutron source in a stainless canister which vertically traverses selected spent fuel (SF) storage cells in the SFP. Four thermal neutron detector tubes in the canister provide continuous

count rate signals to a 4-channel strip recorder. Any gaps in the boraflex result in an increase in counting rate and appear as peaks in the strip chart record.

The licensing basis places some specific constraints on the surveillance. The NRC SER dated February 21, 1990, discusses the 10 full-length boraflex panels to be selected: "10 full-length boraflex panels selected from those that have been exposed to the greatest number of freshly discharged fuel assemblies at the time of the surveillance. Those 10 panels include 4 panels with accelerated exposures and 6 panels selected at random."

The NRC SER dated February 21, 1990, discusses actions to be taken if degraded boraflex is found "...new fuel assemblies or spent fuels with a burn-up less than 38,400 MWD/MT will be stored in a designated area in the fuel storage pool in a checkerboard pattern" Both this stipulation, and the one in the proceeding paragraph, were based on commitments made by WE letters dated April 13, 1989, and November 1, 1989, and referred to in the SER.

Because this non-destructive methodology has been accepted by the NRC for use at PBNP, and because fuel movement will be conducted under normal approved procedures and methods, and, furthermore, because handling of the source, when out of the SFP water, will be conducted utilizing appropriate Health Physics precautions and procedures (to keep exposure ALARA), this testing does not constitute introduction of an unreviewed safety question. (SER 91-062)

80. HP 2.5, (Major), Radiation Work Permit, Revision 18, dated April 12, 1991 (performed April 22, 1991). (Temporary)

The single use temporary procedure change to HP 2.5, Revision 18 allows access to the Unit 1 facade locked gate areas during fuel motion. Access was required to perform acceptance testing of MR 90-161, lead shielding for the fuel transfer tube. HP 2.5 does not permit personnel in high radiation areas during fuel movement through the transfer tube.

Summary of Safety Evaluation: The temporary revision allows personnel involved in acceptance testing of MR 90-161 to remain in the high radiation areas. Work in the HRAs will be controlled by a radiation work permit (RWP). Health Physics coverage will be provided. The testing will be controlled by a RWP as required by both the FSAR and Technical Specifications. (SER 91-034)

81. HP 2.5, (Major), Radiation Work Permit, Revision 19, dated August 23, 1991. (Permanent)

Summary of Safety Evaluation: The change involves the administrative aspects of the RWP system. The FSAR reference involves the RWP applicability to control high radiation area entries. This aspect of the RWP is not affected. The change only affects the administrative use of the RWP and does not result in a reduction in the margin of radiological safety. (SER 91-034-01)

82. ICP 2.1 Appendix A, (Minor), Protection and Safeguards Analog, Revision 9, dated June 7, 1991 (performed November 8, 1991, Unit 1). (Temporary)

ICP 2.7 Appendix A, (Minor), Nuclear Instrumentation Power Range, Revision 2, dated August 23, 1990 (performed November 8, 1991, Unit 1). (Temporary)

Due to voltage spikes on the output of the Unit 1 power range channel N44 (yellow), the yellow channel delta flux controllers will be placed in manual during

performance of ICP 2.1 and ICP 2.7 in order to prevent an inadvertent reactor trip if a spike occurs during this testing. In order to ensure the yellow channel is capable of providing an overtemperature ΔT trip signal if plant conditions warrant, either one or a combination of both delta flux controllers will be adjusted for maximum penalty input to TS ΔT setpoint 1 (ΔT_{sp1}). This maximum penalty equates to 20 mA (which is added to the 10 mA normally seen at the output of the high current selector, 1-TM-404T

Summary of Safety Evaluation: A reactor trip will occur when $\Delta T_{sp1} \leq \Delta T$. At 100% power and no penalty signal applied, the margin to trip is approximately +9°F ($\Delta T_{sp1} > \Delta T$). With the penalty signal applied, the margin to trip is reduced such that power level must be reduced in order to again achieve $\Delta T_{sp1} > \Delta T$. A +6°F margin ($\Delta T_{sp1} > \Delta T$) was selected. A power reduction to approximately 57% achieves this desired margin. Any further reduction in power increases (improves) this margin. (SER 88-132-01)

83. ICP 2.3, (Major), Surveillance Test: Reactor Protection System Logic (Long), Revision 21, dated October 7, 1991. (Permanent)

ICP 2.3 Appendix A, (Minor), S[illegible]veillance Test: Reactor Protection System Logic (Long), Revision 5, dated October 7, 1991. (Permanent)

Summary of Safety Evaluation: The evaluation determines the protection and safety ramifications of simultaneous bypassing of all four NI power range dropped rod circuits during the performance of ICP 2.3. The evaluation concludes that it is considered acceptable to do this because the RPI dropped rod circuit, which is an independent and diverse dropped rod protection control system, provides redundant protective control features.

The protective control actions of the rod drop feature provides a turbine load runback and block rod withdrawal in a dropped rod event. These actions are still provided by the RPI dropped rod protection system, which remains operable while the NI rod drop circuit is bypassed. These features are control system actions versus reactor protection system actions, and as such, will prevent exceeding design fuel limits. They are not relied upon or required to assure safety. Maintaining the RPI dropped rod control protection system operable will ensure that the dropped rod feature will remain operable for accident control as required by FSAR 14.2.3. Also, WCAP-11395A, "Methodology for the Analysis of a Dropped Rod Event," states that no automatic control system features are required for dropped rod protection.

During the performance of ICP 2.3 on either unit, reactor protection is maintained by the opposite train from that which is under test. The change to ICP 2.3 does not have any impact on this feature. (SER 91-084)

84. IT-08A, (Major), Cold Start Testing of Turbine-Driven Auxiliary Feed Pump (Quarterly), Revision 7, dated October 18, 1991 (performed in January 1992, Unit 1). (Temporary)

IT-09A, (Major), Cold Start Testing of Turbine-Driven Auxiliary Feed Pump (Quarterly), Revision 6, dated October 18, 1991 (performed December 10, 1991, Unit 2). (Temporary)

The temporary changes to the inservice tests perform three separate starts of each turbine-driven auxiliary feed pump. The first start will be done from a cold condition and will satisfy the quarterly cold start required by Technical

Specifications. The second start will be done with the governor sensing line disabled and the third start will be done after the governor sensing line is restored as a return to service test. Traces of Terry turbine rpm and pressures will be recorded for each start. The test results will help determine if the governor sensing line can be permanently disabled.

Summary of Safety Evaluation: The test does not affect the ability of the AFW system to deliver water to the steam generators. Only one AFW pump will be out of service while testing. The Technical Specification operability requirements pertaining to the AFW system are met at all times. The pump being tested will be lined up for mini-recirc flow only. The governor sensing line will be disabled by moving the mechanical travel stop on the Fisher regulator so it is not in contact with the governor valve's control linkage. The original position of the travel stop will be recorded and verified; upon test completion it will be returned to its original position and this position verified. (SER 91-114)

85. IT-12, Unit 1, (Major), Component Cooling Water Pumps and Valves (Quarterly), Revision 0, dated June 11, 1991. (New Procedure)

Summary of Safety Evaluation: This test minimizes the possibility of CC system malfunction or damage by administratively controlling maximum flow allowed in the train under test. These are similar to controls required of operators during normal operation and are considered part of the system design. If damage should occur in the train under test, the redundant train is available for safety and operational consideration and TS requirements would be met. This test will not affect cooling to equipment normally supplied by CC at normal operating conditions. (SER 91-049)

86. IT-13, Unit 2, (Major), Component Cooling Water Pumps and Valves (Quarterly), Revision 0, dated June 28, 1991. (New Procedure)

Summary of Safety Evaluation: The test minimizes the possibility of CC system malfunction or damage by administratively controlling maximum flow allowed in train under test. These are similar to controls required of operators during normal operation and are considered part of the system design. If damage should occur in the train under test, the redundant train is available for safety and operational consideration and TS requirements would be met. This test will not affect cooling to equipment normally supplied by CC at normal operating conditions. (SER 91-053)

87. IT-535, Unit 2, (Major), Leakage Reduction and Preventive Maintenance Program Test of Residual Heat Removal System.

Summary of Safety Evaluation: The evaluation analyzes the backleakage of Unit 2 "B" train check valve, 2SI-854B. This is only a concern when the RWST suction MOV 2SI-856B and the containment sump B suction MOV 2SI-854B are open at the same time as in EOP-1.4 when shifting suction of the AUR pump to the containment sump.

The radiological consequences of the backleakage of check valve 2SI-854B is not a factor considering the calculation of SER 89-100-01. The leakage to the RWST is limited by 2SI-856B. The calculation in SER 89-100-01 determines the allowable time to shut 2SI-856B when 2SI-851B valve is open as in EOP-1.4. Using IT-535 U2R17 outage data, 44 minutes would be allowed to shut 2SI-856B. PBNP Technical Specification 15.4.4.1.V requires <2 gph leakage, which will be met by 2SI-856B when it is closed.

The leakage through the check valve is also conservative in that the leakage was measured at 120 psig when the maximum backpressure on this valve would be containment design pressure or 60 psig. (SER 91-094)

88. IT-1190, Part 2A, Unit 1, (Major), Ten-Year Hydrostatic Test of the RHR System (Train A and B), Revision 0, dated March 27, 1991. (New Procedure)

The test hydrostatically tests both trains of the RHR system in accordance with the requirements of ASME Section XI and the PBNP Inservice Test Plan. The through leakage of isolation valve 1SI-852A, as exhibited during U1R17 with the core defueled, prevented the attainment of test pressure in accordance with IT-1190, Part 2 with a small-capacity hydrostatic pump test. Test pressure can be achieved by maintaining RCS pressure greater than or equal to RHR hydro pressure (RCS pressure ≥750 psig).

Summary of Safety Evaluation: Both trains of low head safety injection are isolated and declared out of service for this test. The test is performed with the reactor subcritical and the RCS temperature between 360°F and 500°F and RCS pressure ≥750 psig. This is consistent with TS Section 15.3.1 and 15.3.3 with regard to redundant decay heat removal and low head safety injection requirements.

The basis for TS 15.3.1 states that one reactor coolant pump in operation is required for boron dilution and is sufficient to remove decay heat from FSAR zero power transients (rod withdrawal from subcritical and rod ejection). The basis for TS 15.3.3 states that redundant components of the low head safety injection system may be out of service with RCS temperature between 350°F and 500°F. This is to significantly reduce the decay heat removal requirements of a postulated LOCA and to allow for redundant decay heat removal by the loops and steam generators.

The system will be pressurized to 350 psig for 4 hours and inspected; then the RHR pumps, suction piping and Train B discharge header will be isolated. The remainder of the system will then be pressurized to 750 psig for 4 hours, inspected, pressure reduced to 500 psig and any leakage repaired, and then finally depressurized.

During the test, the reactor coolant loops and steam generators will be used for decay heat removal. Two of two high head SI pumps and at least two of three charging pumps will be available. In addition, both trains of low head safety injection would be available after manual valve realignment. If a leak develops during the hydrostatic test, at least one train of LHSI would still be available after manual valve realignment. (SER 91-021)

89. IT-1190, Part 2B, Unit 1, (Major), Ten-Year Hydrostatic Test of the RHR System (B Train Discharge Header), Revision 0, dated April 26, 1991. (New Procedure)

The test performs a ten-year hydrostatic test of the "B" RHR train discharge piping downstream of 1RH-716B when the RHR system is not required for DHR or LHSI.

Summary of Safety Evaluation: To accommodate the test, a procedural temporary modification will be installed. Relief valve 1SI-861A will be removed and its inlet and discharge connections will be flanged off (RHR/SI piping and PRT). For convenience, this will be done when the cavity is flooded, when one train of RHR DHR may be taken out of service per Technical Specifications.

The RHR train will be placed back into service for DHR operations, without 1SI-861A, to allow mode transition to test conditions. This is acceptable as

1SI-861A is intended for isolated piping relief protection due to thermal expansion considerations. Administrative controls (red tagging 716C and D open) will ensure the piping is not isolated, except for the actual hydro. There will be no functional system impact due to the removal of 1SI-861A.

The hydro will be performed when the RCS loops will be inservice for DHR operations. One RHR/SI train will be available, if needed.

Following the hydro, 1SI-861A will be replaced. This is being done during a condition where containment isolation is required. 1SI-861A is a containment isolation boundary valve. This is acceptable since the outside of containment barrier (RHR being a closed system) will be intact, and the restoration of 1SI-861A will be for a limited duration of time. The blind flange is more than adequate to function as a containment isolation boundary in the interim condition.

Following restoration, the mechanical joints will be inspected for leakage. The testing and recovery will be performed prior to criticality. All Technical Specification requirements for DHR, containment isolation and safeguards equipment availability will be satisfied. This evolution has no impact on LTOP functions. Temporary modification installation and removal is governed by procedure. (SER 91-038)

90. IWP 84-228*A-1 and IWP 84-228*A-2, DC Electrical Distribution. The design package replaces existing swing inverters DY0A and DY0B with new inverters.

Summary of Safety Evaluation: The new inverters are QA and were seismically and environmentally qualified (mild environment). The new inverters have the same rating (10 kVA) as the existing inverters. The new inverters will be seismically mounted in the same locations as the existing inverters. The existing foundations will be modified to accommodate the new inverters. The new inverters have integral static transfer switches which can transfer the instrument buses to an alternate source upon inverter failure. Installation of the alternate source will be addressed by a different design package. The static transfer switches will not be operational until the alternate source is installed. The existing swing inverters supply lighting loads when they are not supplying instrument buses because they cannot be run at zero load. A transfer switch is used to supply lighting from either the inverter or a lighting panel. The new inverters can be run at zero load; therefore, the inverter supply to lighting circuits will be removed. The lighting circuits will be supplied from the lighting panel breakers currently feeding the transfer switch. The transfer switch will be removed. During rewiring of the lighting circuits, temporary lighting will be provided. During the replacement of inverters DY0A and DY0B, the red and blue instrument buses will be fed from the normal inverters (1DY01, 1DY02, 2DY01, 2DY02) and will, therefore, not be affected by the installation. The new inverters have been seismically and environmentally qualified and will be seismically mounted. The new inverters will be mounted in the existing locations, maintaining safety train separation. The new inverters have the same ratings and will function the same as the existing inverters. (SER 91-083)

91. IWP 88-018*D-01, Unit 1, (Minor), Range Change and Calibration of Instrument Loop 1-F00928, Revision 0, dated November 18, 1991. (New Procedure)

IWP 88-019*D-01, Unit 2, (Minor), Range Change and Calibration of Instrument Loop 2-F00928, Revision 0, dated November 18, 1991. (New Procedure)

The IWP controls the range change and calibration of instrument loop 1(2)-F00928, "B Train Low Head Safety Injection," and a meter face changeout of 1(2)-FI-928 and 1(2)-FI-928A. The indicator range will be changed from 0-2000 gpm to 0-2500 gpm. This will be performed during power operation of Unit 1(2). When the instrument loop is out of service for calibration, the meter faces will be changed to reflect the new scale. Standard I&C practices and precautions consistent with those in existing ICPs will be used.

Summary of Safety Evaluation: The instrument loop measures the flow of "B train low head safety injection." It performs an indicating function only and is not required for the safety injection train to be considered operable. During span adjustment and calibration, the transmitter will be isolated from the SI piping. It does not affect any other instrumentation or components which could impact the probability of an accident or malfunction of equipment important to safety occurring.

The range change affects the EOPs. By increasing the range, the calibration accuracy changes from 0.5% to 0.75%. This affects EOPSTPT L.3 which is used to determine if low head SI flow is present. The range change requires changing this setpoint, and procedures EOP-1, Loss of Reactor or Secondary Coolant, and EOP-1.3, Transfer to Containment Sump Recirculation. The setpoint change does not affect the conservatism of these procedures, and that the range change selected for the instrument is sufficient to adequately determine low head SI flow. Therefore, the consequences of any accident or malfunction of equipment important to safety will not be increased. (SER 91-074-04)

92. IWP 88-018*D-03, Unit 1, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 1-F00962, Revision 0, dated November 18, 1991. (New Procedure)

IWP 88-019*D-03, Unit 2, (Minor), Range Change, Sensing Capsule Replacement, and Calibration of Instrument Loop 2-F00962, Revision 0, dated November 18, 1991. (New Procedure)

Summary of Safety Evaluation: The IWPs control the range change, sensing capsule replacement, and calibration of instrument loop 1(2)-F00962, "A Train Containment Spray Flow," and the meter face changeout of 1(2)-FI-962. The range change involves replacing the transmitter sensing capsule to allow the transmitter span to be increased. This will allow the indicating range to be changed from 0-1320 gpm to 0-1800 gpm. This will be done during power operation of Unit 1(2). When the instrument loop is out of service for capsule replacement and calibration, the meter face will be changed out to reflect the new scale. Standard I&C practices and precautions consistent with those in existing ICPs will be used.

The instrument loop measures the flow of "A" train containment spray. It performs an indicating function only and is not required for the containment spray train to be considered operable. It does not affect any other instrumentation or components which could impact the probability of an accident or malfunction of equipment important to safety occurring.

During the sensor capsule changeout and calibration, the transmitter will be isolated from the spray piping. The filling and venting of the instrument will be done during IT-05(06), the monthly tests of the containment spray system. This is done so containment integrity is maintained during the filling of the lines because this cannot be done while the instrument is isolated from the spray piping. During IT-05(06), containment isolation is provided by closed isolation valve 1(2)SI-868A.

Other instrumentation such as spray pump pressure, spray additive flow and tank level, and containment temperature and pressure will be available to indicate containment spray initiation. Therefore, the consequences of an accident or malfunction of equipment important to safety will not be increased. (SER 91-027-05)

93. IWP 88-018*D-04, Unit 1, (Minor), Range Change, Sensing Line Capsule Replacement, and Calibration of Instrument Loop 1-F00963, Revision 0, dated November 18, 1991. (New Procedure)

IWP 88-019*D-04, Unit 1, (Minor), Range Change, Sensing Line Capsule Replacement, and Calibration of Instrument Loop 2-F00963, Revision 0, dated November 18, 1991. (New Procedure)

Summary of Safety Evaluation: The IWPs control the range change, sensing capsule replacement, and calibration of instrument loop 1(2)-F00963, "B Train Containment Spray Flow," and the meter face changeout of 1(2)-FI-963. The range change involves replacing the transmitter sensing capsule to allow the transmitter span to be increased. This will allow the indicating range to be changed from 0-1320 gpm to 0-1800 gpm. This will be done during power operation of Unit 1(2). When the instrument loop is out of service for capsule replacement and calibration, the meter face will be changed out to reflect the new scale. Standard I&C practices and precautions consistent with those in existing ICPs will be used.

The instrument loop measures the flow of "B" train containment spray. It performs an indicating function only and is not required for the containment spray train to be considered operable. It does not affect any other instrumentation or components which could impact the probability of an accident or malfunction of equipment important to safety occurring.

During the sensor capsule changeout and calibration, the transmitter will be isolated from the spray piping. The filling and venting of the instrument will be done during IT-05(06), the monthly tests of the containment spray system. This is done so containment integrity is maintained during the filling of the lines because this cannot be done while the instrument is isolated from the spray piping. During IT-05(06), containment isolation is provided by closed isolation valve 1(2)SI-868B. Other instrumentation such as spray pump pressure, spray additive flow and tank level, and containment temperature and pressure will be available to indicate containment spray initiation. Therefore, the consequences of an accident or malfunction of equipment important to safety will not be increased. (SER 91-027-06)

94. IWP 88-020-1P10A, Unit 1, (Minor), P-010A Modified Rotating Assembly Installation, Revision 0, dated September 23, 1991. (New Procedure)

IWP 88-020-1P10B, Unit 1, (Minor), P-010B Modified Rotating Assembly Installation, Revision 0, dated September 23, 1991. (New Procedure)

IWP 88-020-2P10A, Unit 2, (Minor), P-010A Modified Rotating Assembly, Revision 0, dated September 23, 1991. (New Procedure)

IWP 88-020-2P10B, Unit 2, (Minor), P-010B Modified Rotating Assembly, Revision 0, dated September 23, 1991. (New Procedure)

Summary of Safety Evaluation: MR 88-020 replaces the RHR pump thrust bearings and arranges them in a way to avoid ball skidding and bearing noise which is

possible in the original arrangement. It replaces several thrust bearing retaining parts to result in a bearing capture method which lends itself to obtaining and maintaining proper axial end float of the shaft. It also replaces the existing John Crane Type 1 seal with a John Crane Type 1B seal. The design was done by the original equipment manufacturer. The specification required the design also satisfy the original Westinghouse equipment specification. Based on these requirements, the design is such that the capability of the RHR pump to function in its designed capacity is not degraded.

To verify the design of this modification, the first assembly to be installed will be subjected to the following tests. Hydrostatic test to verify seal integrity and leak tightness; pump performance test at various flow rates at which time vibration signatures will be obtained to verify acceptability; and seal water heat exchanger performance testing to verify adequate cooling is provided for the seal

If the testing results are satisfactory for the first assembly installed, thereby qualifying the design, the remaining pumps may have modified rotating assemblies installed and accepted with relaxed testing requirements.

The changes to the pump essentially do not change its performance characteristics but do enhance its maintenance aspects and pressure rating (of the seal). Since the noted changes do not change the structural integrity of the pump, there is no impact to the original seismic qualifications. The materials of construction for the replacement parts are essentially the same as the original. (SER 91-063)

95. IWP 88-098*A1, Unit 2, (Minor), SI/CS Test Line--Mechanical Work not Requiring an Outage, Revision 0, dated September 13, 1991. (New Procedure)

IWP 88-098*B1, Unit 2, (Minor), RHR Test Line--Mechanical Work not Requiring an Outage, Revision 0, dated September 13, 1991. (New Procedure)

Summary of Safety Evaluation: The IWPs install piping and supports for the full flow test lines for the SI, containment spray, and RHR systems. The piping and supports are installed while both units are at power so work during the outage can proceed on schedule. There will be no impacts to existing systems under this design package and IWPs. All system tie-ins and modifications to existing system supports are performed during U2R17.

Construction under this design package and IWPs is done in accordance with B31.1-1967. The installation uses standard rigging practices for erection of piping. To assure that the interim configurations are adequate from a Seismic 2/1 concern and that there is no effect on the existing systems 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 sequence of installation with regard to piping and supports is determined at the time of installation, but is done in accordance with the method outlined above. Also, the rigging uses a minimum safety factor of five beyond rated capacity and prevents any possible 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. Based on these controls, it is concluded that there is no effect on the adjacent systems as a result of this installation. (SER 91-074)

IWP 88-098*A2, Unit 2, (Minor), SI/CS Test Line--Mechanical Work to be Performed During U2R17, Revision 0, dated October 8, 1991. (Permanent)

IWP 88-098*B2, Unit 2, (Minor), RHR Test Line--Mechanical Work to be Performed During U2R17, Revision 0, dated October 4, 1991. (Permanent)

The procedures install the tie-ins for the full flow test lines for the Unit 2 SI, CS and RHR systems as documented in MR 88-098*A and MR 88-098*B. IWP 88-092*A2 implements a procedural temporary modification.

IWP 88-098*A2 performs the tie-in of the test line to the discharge header of the SI pumps to the discharge header of the containment spray pumps and ties in the return line to the RWST. IWP 88-098*B2 performs the tie-in of the RHR test line to the discharge headers of the RHR pumps, and ties in the test line to valve 2RH-742, where the test line flow can be returned to the RWST.

Summary of Safety Evaluation: All of the work performed under these two IWPs is performed with Unit 2 in the cold or refueling condition, with the exception of RHR, which is installed with the core off-loaded The affected systems are not required to be available to perform their safety function during the plant conditions under which the tie-ins are being made.

In addition to performing the work during these plant conditions, deliberate precautions are taken during construction of the tie-ins to assure that the construction has no impact on operating safety-related equipment in the area. These 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 are reviewed and documented by engineering personnel on a shiftly basis.

In addition to these controls, all post-modification functional testing is performed prior to placing the test line into service. These tests include performance of the monthly pump performance inservice test and the inservice test which tests flow to the core. It also includes NDE and hydrostatic testing per original Westinghouse specifications and B31.1-1967.

IWP 88-098*A2; Tie-in of SI and Containment Spray: The tie-in to the SI system is performed during cold or refueling shutdown and non-reduced inventory operation, with the SI system out of service.

The tie-in to the containment spray system is performed during cold or refueling shutdown, with the CS system out of service. The manual gate valve to the containment (868 A and B) is closed for containment closure concerns.

The tie-in to the RWST is done during the cavity-flooded phase. The RWST is tagged out of service. In order to properly drain and inspect the RWST, a tank entry is made. This requires opening the lower manway. To assure adequate ventilation, a procedural temporary modification is done to install a temporary blower and duct which vents the top of the RWST to the purge exhaust line. This sends the vented air through an existing filter bank and radiation monitor. This is supplemented by tritium sampling by Chemistry. The temporary modification also documents a pump and jumper hose connected to the RWST drain valve which pumps water to a local drain (2CV-D-19). This is installed to drain the last bit of water from the RWST (about 12"). The water is routed to a CVCS holdup tank and returned to the RWST after the tank entry is complete. After the line is used, a DI

water flush is made in the tank and in the pipe routing used for the RWST pump down. The temporary pump being used has a shutoff head of 56' (relatively low pressure). Also, an engineering review is performed and documented in the IWP prior to starting the draindown.

IWP 88-098*B2; Tie-in of RHR: The installation of these tie-ins is performed with both trains out of service with no fuel in the core. The initial return-to-service check of the pumps is IT-04A, which is performed with pump suction from the RCS loops. This is necessary because the RWST is empty. After the return-to-service test, the fuel may be returned to the core and the cavity can be subsequently restored. Functional test of the test line and performance of the flow test to the core is performed prior to accepting the test line installation. The functional tests are performed such that one train of RHR will be in service at all times to meet TS 15.3.8.

Based on the above, it is concluded that there are no unreviewed safety questions associated with these IWPs. (SER 91-074-01)

97. IWP 88-098*A2, (Minor), Installation of Unit 2 RHR, SI and CS Test Lines, Revision 1, dated October 8, 1991. (Permanent)

Summary of Safety Evaluation: This amended SER allows core reload with only one train of RHR operable.

General Considerations for Both IWPs: All of the work performed under these two IWPs is performed during the plant conditions which allow the affected system or train to be out of service.

IWP 88-098*A2; Tie-in of SI and Containment Spray: The tie-in to the containment spray system is performed during cold or refueling shutdown, with the CS system out of service. The manual gate valve to the containment (868 A and B) is closed for containment closure concerns when containment closure is required.

IWP 88-098*B2; Tie-in of RHR: The core is off-loaded prior to making the tie-ins to the RHR system. The RHR "A" train tie-in through the test line root valve is completed, this new root valve is tagged shut; and all post-installation testing required to prove operability of the "A" train for decay heat removal is completed before core reload begins. This testing consists of NDE and pressure testing of the new components up to the root valve and the performance of the RHR "A" train pump test, IT-04A. The seismic integrity of the "A" train during this interim condition is maintained by having the first permanent support downstream of this root valve installed along with the piping between the root valve and this support in addition to installing two temporary supports in the vicinity of the root valve. The seismic acceptability of this arrangement is documented in the procedural temporary modification written for the temporary supports. Installation and testing of the remainder of the RHR test line and testing of the "B" train of RHR to prove operability for decay heat removal is completed prior to lowering the refueling cavity water level. Flow testing to the core and through the test line along with NUREG-0578 leak testing is completed prior to final acceptance of the modification and Unit 2 criticality. The IWP is written such that at least one train of RHR is operable at all times that fuel is in the core. The Technical Specification requirements pertaining to refueling are met at all times. (SER 91-074-05)

98. IWP 88-099*C, Unit 2, (Minor), 2P29 Auxiliary Feedwater Pump Recirc Line Improvement, Revision 0, dated October 4, 1991. (New Procedure)

MR 88-099 replaces the existing mini-recirc lines for the auxiliary feedwater (AFW) pumps with new larger capacity mini-recirc lines. The flow rate is increased to protect the pump from the adverse effects of hydraulic instability at low flow rates. The modification also adds recirc flow measurement instrumentation to provide local flow indication for inservice testing of the AFW pumps. MR 88-099 was initiated in response to NRC Bulletin 88-04 with refinements added by NRC Generic Letter 89-04. MR 88-099*C controls installation for the Unit 2 steam-driven auxiliary feedwater pump (2P29).

Summary of Safety Evaluation: The capacity of the mini-recirc lines for 2P29 is increased from 30 gpm to a minimum of 100 gpm based on the recommendation of the manufacturer. The increased flow through the mini-recirc line protects the pumps from the adverse effects of hydraulic instability at low flow rates. Calculation N-91-032 analyzes the impact of the increased mini-recirc flow if 2AF-4002 sticks open. The calculation shows that the pump will still deliver 162 gpm vice the 200 gpm nominal flow to each of the Unit 2 steam generators. If single failure criteria is applied to the safety-related components of the AFW system and the single failure is 2AF-4002 sticking open, 1P29 at 400 gpm, P38A and P38B at 200 gpm each and 2P29 at 324 gpm provide adequate flow to the steam generators for decay heat removal. The new malfunction is within the scope of the analysis for the normal loss of feedwater accident. This single failure also causes the pump's operating point to move further out on the pump curve, closer to pump runout conditions ~450-500 gpm. This operating point is well within the capabilities of the pump. The primary concern in this condition, however, is to ensure NSPH requirements are met. The NPSH required for 500 gpm is less than the NPSH available at minimum CST level (4.5'). 2AF-4002 is included in the ASME Section XI test program and is verified to operate properly by IT-295.

This modification is completed with Unit 1 at power and Unit 2 shut down. With its associated unit shut down, TS 15.3.4.C allows a steam-driven AFW pump to be taken out of service for maintenance or testing for an indefinite period of time. Because this is within the time constraints of the TS, there is no reduction in the margin of safety.

The AFW system is a Seismic Class 1 system. The mini-recirc piping and supports for the modification are designed to meet the Class 1 requirements. During installation, the only piping and supports affected are for the portions of the mini-recirc lines being worked at that time. The inservice AFW pumps and their associated suction, discharge and mini-recirc piping are still considered to meet Seismic Class 1 requirements and be operable to perform their intended safety function. (SER 91-0 5-04)

99. IWP 90-191-1, (Minor), Correction of DC Power Distribution Deficiency in C01, Revision 0, dated April 17, 1991. (New Procedure)

The deficiency on the supply of MOBs 155 through 164 will be eliminated by relocating wires from the load side to the supply side of MOB-100. To maintain power to these MOBs during this change, temporary connections will be made from the load side of MOB-106 to the load side of spare breaker MOB-156, and power will be backfed through MOB-156 to supply the set of MOBs.

Summary of Safety Evaluation: It is expected that the procedure will be performed with Unit 2 at power. MOB-155 supplies an alarm circuit and MOB-157 provides

only indication. If deenergized the containment isolation valves supplied by MOB-157,159,160,161,and 162 would all fail closed. For valves 2CV-538, 2CV-897A, and 2CV-1723 this would cause no immediate safety or operability concerns. For 2CV-3200A&C, this would result in the loss of the R211/212 containment monitoring system. For 2CV-3047 one of the two instrument air supplies to the Unit 2 containment would be lost. Instrument air to containment should remain available through 2CV-3048. The circuits supplied by MOB-158, 163, and 164 supply safeguards functions. The loss of safeguards power will disable the automatic function. No action will occur except that the following annunciators will alarm:

CO2F-4-3	Unit 2 Safeguard DC Control Power Failure
C01A-2-8	Unit 1 AFWS Disabled
C01A-2-10	Unit 2 AFWS Disabled
C02F-4-4	Unit 2 Common Critical Power Failure

The likelihood of safeguards power loss is remote. If it did occur it should not result in any immediate challenge to the continued operation of Unit 2. The failure would be immediately detected and quickly corrected. The circuit change is considered to be an improvement to the safety and reliability of the plant in that it will eliminate the possibility of a single fault disabling multiple safety circuits. (SER 91-029-01)

100. IWP 90-228, Unit 2, 4160V Breaker Control Power Fuse Modifications, Revision 0, dated September 13, 1991. (New Procedure)

Summary of Safety Evaluation: The design package involves the replacement of the existing swing inverters for the red and blue instrument buses. During the replacement of inverters DY0A and DY0B, the red and blue instrument buses will be fed from the normal inverters (1DY01, 1DY02, 2DY01, 2DY02) and will, therefore, not be affected by the installation. The new inverters have been seismically and environmentally qualified and will be seismically mounted. The new inverters will be mounted in the existing locations, maintaining safety train separation. The new inverters have the same ratings and will function the same as the existing inverters. (SER 91-075)

101. OI-35, (Minor), Electrical Equipment Operation, Revision 9, dated February 1, 1991. (Permanent)

Summary of Safety Evaluation: The new alternate shutdown load centers, B08 and B09, are not discussed in the FSAR. The new guidance for the breaker operation in these load centers (new Steps 4.7 - 4.15) are similar in nature to the discussions of operation of other breakers in Section 4.0, "Instructions for Racking in/Racking Out 480 V Breakers." Notes and precautions relative to breaker operation, safety, and administrative controls (involvement/permission of the DSS), have been added to the specific steps, as well as to Section 2.0, "Precautions and Limitations."

Section 12 of OI-35 discusses the 86 lockout relays. Prior to this revision, the section only discussed 4160 V/480 V lockouts ("standard practice is to not reset any 86 lockout relays except under the administrative control of the DSS ..."). This section is being expanded to include 345 kV/13.8 kV lockouts. The same standard practices discussed above is also applicable, with this change, to the 345 kV/13.8 kV lockouts, with some additional guidance that the system control supervisor should be consulted by the DSS. The change is a conservative measure and is indicative of good operational controls.

Section 15.0, "Operation of B03/B01 and/or B04/B02 Tie Breakers," was added. It responds to an internal nonconformance report which documents an electrical separation discrepancy for safety-related cables, and postulate a single failure sequence which could: (1) cause both bus tie breakers to close simultaneously, or (2) lose the ability to strip the nonsafety-related buses (B01 and B02) from B03 and B04; which would result in an overload of both emergency diesel generators. Corrective Action 4 directs Operations to "Revise OI-35 to include controls for closing the B03/B01 and B04/B02 bus tie breakers." (SER 91-004)

102. OP-3A, (Major), Normal Power Operation to Low Power Operation, Revision 28, dated April 3, 1991. (Permanent)

The change adds a step, which directs auxiliary feedwater system inservice testing to be performed at a time during a power reduction when unit loading is between 20% and 30% power and the unit is approaching end of life (RCS boron ≤200 ppm). This test was previously performed when reactor power was <2% and Tavg was controlled to 540°F using pressure control of the steam dump and manual control of the rod control system.

Summary of Safety Evaluation: The change was evaluated from two perspectives: Power transient; and thermal fatigue of the feedwater nozzles to the steam generator.

Power Transient: The change in operation was proposed by the U1R17 outage critique with the reasoning that performing the test after the reactor is subcritical creates a potential restart accident due to decreasing temperatures. At EOL, it is difficult to maintain the reactor critical due to the xenon buildup occurring as power is reduced in the course of the procedure. Performing IT-290 adds cold auxiliary feedwater from the turbine-driven auxiliary feedwater pump at full flow to the steam generators.

The amount of positive reactivity addition will be the same whether the test is performed at <2% power (where it is presently specified to perform the test) or at 20-25% power (the proposed power level at which the test will be conducted). What will be enhanced at the higher power level is the ability to control reactor power with the plant steaming to the turbine, in contrast to the low power level with steam dumps in operation controlling to set pressure.

The transient is not "analyzed," per se, in the FSAR. Similar transients, however, are analyzed. The most similar is FSAR Section 14.1.6, "Reduction in Feedwater Enthalpy." The postulated transient results from the bypass of main feedwater from some of the feedwater heaters resulting in a feedwater temperature decrease of 15°F. From an initial conduction of 100% power, a reactor power increase of ~9% is seen.

For the transient, assuming reactor power initially at 25%, a calculation shows a decrease in feedwater temperature (when the auxiliary feedwater at an assumed 70°F is mixed with main feedwater at 323°F) of 25°F from 323°F to 298°F.

Because the transient is at 25% power, feed flow is also ~25% rated. Although the 25°F decrease is greater than the 15°F decrease of FSAR 14.1.6, the analyzed case is bounding because it occurs at a feed flow ~4 times as great -- the greater reactivity addition occurs here. The analyzed transient occurs at initial power level of 100% as well.

Thermal Fatigue: The Unit 2 steam generator stress report was reviewed. The fatigue evaluation for the generator feedwater nozzles is based on the specification that the nozzle must survive 25,000 cycles of adding cold feedwater (70°F) to the generators from an initial hot standby, no-load condition of 547°F at a rate of 400 gpm. The evaluation uses this transient as the most limiting fatigue cycle the nozzles will see, and shows acceptability in meeting the specification.

OP-3A allows a recreation of the above-described cycle by performing IT-290 at hot, no-load conditions. Cold auxiliary feedwater at rated flow is inserted to the generator past the feed nozzle (with integral thermal sleeve), and one of those cycles is used. Some main feed flow may exist during the test, but the amount is small and not subject to preheating as the point in the procedure for auxiliary feedwater test performance is after feedwater heaters are secured.

The revision to OP-3A by conducting the test at 20-25% power, allows for a less severe thermal cycle to the feedwater nozzle by subjecting the nozzle to a smaller temperature difference (ΔT). An expected feedwater temperature of ~300°F will now pass a feedwater nozzle which will be at a lower temperature (~534°F according to secondary plant design heat balance for 25% power). Because the change does not propose any change in test frequency, and because the ΔT the nozzle will experience is much less than previous, the change is conservative and poses no new questions regarding thermal fatigue. (SER 91-022)

103. RDW 17.3, (Minor), Processing Bead Resin by Dewatering, Revision 2, dated December 4, 1991. (Permanent)

The procedure was expanded to include setup and operation of dewatering equipment. Concerns and monitoring guidelines for potential problems associated with the introduction of strong oxidizing agents such as nitric acid and the formation of methane/explosive gas were addressed in response to NRC IN 90-050.

Summary of Safety Evaluation: Nitric acid is the major strong oxidizing agent of concern. The nitrates constituent is required to produce the noted exothermic reaction. PBNP has a very low probability of introduction of this reagent into the resin shipping container. However, the procedure does adequately addresses required actions. The temperature of the resin is continuously monitored throughout container loading and the dewatering process by remote means. Upon observance of an exothermic reaction, the operator is directed to flood the container with water to cool, dilute and cease the reaction. (Nitric acid is not used on-site in bulk quantities; the procedure caution prevents introduction of the material into the resin.)

Methane gas is generated via biological degradation of the organic resin media. In general, this is not a problem throughout the industry; however, it has been observed on a few occasions following resin dewatering of resin beds contaminated either by sewage or, in a few cases, by floor drains. PBNP uses resins in the primary systems and not for wastewater processing. The probability of biological agent contamination of the resin is very low. The procedure requires explosive gas monitoring at the disposal container vent following dewatering and daily thereafter, until the container is shipped to a disposal site. (SER 91-110)

104. RESP 1.1, (Major), Rod Control System: Rod Drop Testing, Revision 2, dated October 18, 1990 (performed September 23, 1991, Unit 2). (Temporary)

Summary of Safety Evaluation: The temporary change connects test equipment that is partially installed per IWP 87-139*B to measure control rod drive mechanism

coils. The equipment was installed in the Unit 2 rod drive cabinets but were not electrically tied in to minimize the chances of causing rod drops during the installation at full power. A temporary tie-in will be made for three of the 33 control rods after reaching zero power at end of life (EOL) for Unit 2 Cycle 17. This will allow an operational checkout of the new test equipment during the EOL rod drop test before installation resumes during the refueling shutdown to make the permanent tie-in. The new test equipment will be used in parallel with the standard test equipment (visicorder). (SER 91-082)

105. RESP 1.3, (Major), Multi-Map Calibration of Nuclear Power Range Detectors, Revision 3, dated May 8, 1990 (performed December 13, 1991, Unit 2). (Temporary)

Summary of Safety Evaluation: The temporary change allows axial offset constants, which are conservative with respect to the constants determined by the recent multi-map calibration, to remain in effect of present value. These constants reflect the gain of the summing amplifiers which generate the Δflux input to the OTΔT setpoint generating instruments. Leaving the more conservative (i.e., larger) constants in place will result in a more pronounced Δflux effect on OTΔT and will cause a more conservative (i.e., lower) OTΔT to be generated. It is not desirable to use the axial offset constants determined for N42, N43 or N44 because the change would be in the nonconservative direction and a review of the acceptance criteria and methodology of determining setpoints is desirable before the three constants would be entered on the three channels. Changes in sequence for inputting PPCS constants is irrelevant with regard to safety concerns. (The change is also conservative with respect to delta flux indication on the main control boards.) (SER 91-117)

106. RESP 4.1, (Major), Initial Criticality and ARO Physics Test, Revision 5, dated May 14, 1991 (performed October 23, 1991, Unit 2). (Temporary)

The physics test was changed to add a dynamic method of measuring control rod worth. The new method is described in Westinghouse safety evaluation SECL 91-138.

Summary of Safety Evaluation: The procedure change allows a test to be performed which is not described in the FSAR. The test performs dynamic rod worth measurement (DRWM) during the Unit 2 zero-power physics testing time interval. The data collected from the test is used as proof-of-principle verification of the testing technique and may be used in a topical report to the NRC at a later date. The DRWM results are not used to verify conformance to the plant Technical Specifications regarding rod worth or shutdown margin. The existing rod worth measurement testing method will be used to verify rod worth as has been done in previous reloads.

The DRWM test limits the excess core reactivity to 75 pcm which results in a stable reactor period of 80-90 seconds. This flux transient will be substantially less severe than those predicted in the FSAR analysis on uncontrolled rod withdrawal which assumes two control banks are moving simultaneously with full overlap and at the maximum speed of 72 steps per minute. Additionally, even if no operator action was taken, the RPS will remain capable of terminating the transient before any licensing basis criteria would be violated. The 75 pcm reactivity addition is also bounded by the present BOL physics testing procedure which calls for as much as 80 pcm reactivity addition.

LOCA and SGTR related analysis are not adversely affected by this test.

The evaluation examines the effect of increasing rod speed to 72 steps per minute and concludes that there will be no detrimental effect on the rod drive mechanisms. The current BOL physics test already allows 72 steps per minute rod speed; thus, there is no change in the allowed rod speed.

The existing physics test Technical Specification exemptions are sufficient to allow this testing to take place. The DRWM test does not involve an unreviewed safety question. (SER 91-099)

107. RMP 29A, Unit 1, (Minor), 1A01, 1A02, 1X11 and 1X12 Breaker, Bus and Transformer Maintenance, Revision 0, dated February 1, 1991. (New Procedure)

RMP 29B, Unit 2, (Minor), 2A01, 2A02, 2X11 and 2X12 Breaker, Bus and Transformer Maintenance, Revision 0, dated February 1, 1991. (New Procedure)

RMP 23F, Unit 1, (Minor), 480V Breaker Maintenance, Revision 0, dated February 1, 1991. (New Procedure)

RMP 23G, Unit 2, (Minor), 480 V Breaker Maintenance, Revision 0, dated February 1, 1991. (New Procedure)

Summary of Safety Evaluation: RMP 23F (Unit 1) and RMP 23G (Unit 2) controls use of the bus tie breakers to support normal bus supply breaker maintenance. RMP 29A (Unit 1) and RMP 29B (Unit 2) controls use of the bus tie breakers to support maintenance of buses A01 and A02, transformers X11 and X12 and breakers adjacent to these buses.

The RMPs assure that the single failure condition is eliminated by the following:

- Before placing control power fuses for B03/B01 or B04/B02 bus tie breakers in the ON position, the opposite train diesel is verified operable to ensure a single failure fault in the control circuit of the bus tie breaker will not potentially place the only operable diesel in an unanalyzed condition.

- Immediately after the bus tie breaker is closed, the control wire from (+)DC to the "x" coil is lifted and taped at the bus tie breaker cubicle. This isolates the "x" coil from control cabling that runs in the same tray as control cabling for the opposite train bus tie breaker.

- The evolution of the action steps in these RMPs assures that once the control power fuses to a tie breaker are installed, the procedure will continue through lifting of the (+)DC control wire to the "x" coil of that breaker before beginning use of the opposite train tie breaker. Conversely, restoration of the lifted control lead will be completed immediately after placing the breaker control power fuses in the OFF position and before beginning use of the opposite train tie breaker.

Use of the B03/B01 and B04/B02 bus tie breakers, as controlled by the RMPs, bypasses the 480 V system dead bus transfer scheme. Bypass is established by installing jumper wires in the breaker control scheme. However, the RMPs procedurally verify that buses B03 and B01 (B04 and B02) are supplied by the same offsite power source immediately before bus tie breaker use or restoration of normal supply breakers. With this condition met, no phase difference will exist and no synchronization is required to tie the buses together. Further, the RMPs remove

the bypass jumpers after the tie breakers are closed or removed from service so that the dead bus transfer scheme is not permanently compromised. (SER 89-059-04)

108. RMP 29E, Unit 2, (Minor), 2A05 Breaker and Bus Maintenance, Revision 0, dated October 11, 1991. (New Procedure)

RMP 29E provides procedural guidance to electrically isolate and restore Unit 2 4160 V safeguards bus 2A05 to allow bus/breaker maintenance on 2A05, 2A52-76 and 2A52-75. This will require that 480 V safeguards bus 2B03 be tied to 2B04 for the duration of RMP maintenance.

Summary of Safety Evaluation: The required Unit 2 plant condition is core defueled. In this condition, unit specific safeguards on bus 2B03 may be stripped. If needed, Unit 2 "B" train loads will be placed in service on 2B04 before the "A" train loads are stripped from 2B03. (SER 91-087-01)

109. RMP 29F, Unit 2, (Minor), 2A06 Breaker and Bus Maintenance, Revision 0, dated October 11, 1991. (New Procedure)

RMP 29F provides procedural guidance to electrically isolate and restore Unit 2 4160 V safeguards bus 2A06 to allow bus/breaker maintenance on 2A06, 2A52-70 and 2A52-69. This will require that 480 V safeguards bus 2B04 be tied to 2B03 for the duration of maintenance on the 2A06 bus.

Summary of Safety Evaluation: The required Unit 2 plant condition is core defueled. In this condition, unit specific safeguards on bus 2B04 may be stripped. If needed, Unit 2 "A" train loads will be placed in service on 2B03 before the "B" train loads are stripped from 2B04. (SER 91-087)

110. RMP 47A, Unit 2, (Minor), 2X03 Maintenance, Revision 0, dated October 16, 1991. (New Procedure)

Summary of Safety Evaluation: The procedure controls the removal from and return to service of 2X03, the high voltage station auxiliary transformer, and associated breakers and equipment, for maintenance. During the time 2X03 is removed from service, offsite power is provided to both units through 1X03. When 2X03 is being removed from and returned to service, 1X03 and 2X03 will be paralleled for a short period of time. (SER 91-095)

111. RMP 71, Unit 1, (Minor) A Train Degraded/Loss of Voltage Test, Revision 8, dated November 7, 1990, and RMP 73, Unit 2, (Minor), A Train Degraded/Loss of Voltage Relay Tests, Revision 5, dated October 26, 1990, were temporarily changed to allow performance of "A" train undervoltage relay testing with EDG fuel oil supply pump P70B out of service.

Summary of Safety Evaluation: With P70B out of service, P70A is left to supply fuel oil G01 and G02 emergency diesel generators. The undervoltage relay testing segments last approximately 1 hour per segment. If power was lost to P70A during testing, the diesel generators would still have an ample supply of fuel oil in their engine mounted sumps and associated local day tanks to allow sufficient time to recover from undervoltage relay testing. There would be positive control throughout the undervoltage testing since the maintenance personnel performing the testing will be in direct communication with the control room.

Testing of the undervoltage relays does not take 1A05 and 2A05 out of service. G01 still starts and provides power to P70A. The testing only takes one channel of undervoltage protection out of service at a time. Should an actuation of an undervoltage occur, G01 will start, pick up the bus and supply P70A, as would occur during an actual event. The potential for losing power to P70A is not increased by the performance of this testing. (SER 91-001)

112. RMP 75, Unit 1, 4 kV Loss of Voltage Relaying and Auxiliary Feedwater Initiation Test, Revision 2, dated June 28, 1991. (Permanent)

Summary of Safety Evaluation: The change adds surveillance for undervoltage relay testing associated with nonvital bus (A01 and A02) stripping, including reactor coolant pump (RCP) breaker trip on bus undervoltage. The revision to this procedure is the Unit 1 complement of RMP 76, Revision 2 which was evaluated under SER 90-079. The results and conclusions of that SER remain valid for this procedure revision. (SER 90-079-01)

113. RMP 172, (Minor), Monitor Emergency Diesel Generator Fast Start Voltage and Breaker Closure, Revision 0, dated September 24, 1991 (New Procedure)

The procedure directs monitoring of emergency diesel generator output voltage from the time of fast start until output breaker closure. The purpose of monitoring is to verify that diesel output is up to normal when the output breaker closes following a fast start. There is no close permissive interlock which checks output voltage prior to the output breakers closing. This monitoring is a commitment to the NRC resulting from EDSFI finding 90-201-13.

Summary of Safety Evaluation: The trigger and breaker closure monitoring contacts use spare contacts for their actuation signal. Because these contacts are not electrically connected to any diesel generator control circuit, the monitoring can have no effect on diesel generator operation. The leads used to monitor output voltage are fused so that any significant short circuit would cause the fuse to blow, and would not affect the operation of the diesel potential transformer or voltage regulator. The monitoring function itself, therefore, has no effect on diesel operability, and both diesel generators may be monitored simultaneously without an operability concern. (SER 91-085)

114. SMP 1045, Unit 1, (Minor), Replace Flange and Reducer Downstream of Valve SW-144, Revision 1, dated April 11, 1991. (Permanent)

The 8"x14" reducer downstream of SW-144 eroded due to cavitation. Per TM 89-033, a second reducer was sandwiched over the top of the original and welded in place to provide replacement wall thickness while the line was in service. MWR 900299 was written to replace the reducer and flange with stainless steel components. The material substitution is authorized based on a generic evaluation done for MR 87-158.

Summary of Safety Evaluation: The work requires the Unit 1 fan coolers to be out of service. Thus, the replacement is done during a refueling or cold shutdown. Other equipment to be taken out of service is addressed. This includes the service building cooling coil, service building chiller, Unit 1 electrical equipment room cooler, auxiliary building cooling coil, SGBD analysis cabinet, and the gas analyzer sample cooler. Outside air can be recirculated as necessary in those areas where ventilation cooling coils are taken out of service. In regard to the gas analyzer sample cooler, the RETS-required sample from the gas decay tanks is not affected since this sample is dry gas and no moisture needs to be condensed. Other

administratively- required samples tal:e which do need the cooler are rescheduled with forewarning given to Chemistry.

Replacement of the reducer downstream of SW-144 results in opening an unisolable 14" hole in service water overboard. The location at which the 14" line connects into the 20" overboard return header is at El. 34'14" south of the 20" flange (used for dam installation) east of C59. Thus, the level of the water in the service water overboard lines has to be lowered to an elevation below the north/south horizontal run to 20" pipe (in which the 20" dam flange is installed). To minimize effects on Unit 2 (at 100% power), the dam is installed in the 20" header east of C59. This minimizes the amount of water that will be able to return to the Unit 1 side of the return header from the CC heat exchangers (which tie in north of the dam) while the 14" line is cut. The return header level is controlled via the butterfly valves (SW-146 and SW-104) at the entrance to the Units 1 and 2 circulating water system. Pressure indicators are positioned at each valve location for monitoring head in the return headers. Two valve bonnets at the PAB coolers service water return piping are removed to provide a high point vent (located above El. 66' in Unit 1 fan room) to help ensure good water head indication at SW-146 and SW-104. (SER 90-065-01)

115. SMP 1066, (Minor), Shipment of the Reactor Vessel Surveillance Specimen Capsule, Revision 0, dated January 8, 1991. (Permanent)

Surveillance Capsule S from the Unit 2 reactor vessel was removed from the vessel during the previous outage and is currently stored at location C-30 in the spent fuel pool (SFP). The purpose of this special evolution is to transfer the capsule to a shipping cask for shipment to an off-site facility for testing. This will involve bringing in a cask which weighs approximately 11 tons, moving the cask over the SFP to a laydown area for preparation, moving the cask over the cask loading area of the SFP, lowering basket from the cask into the SFP, loading the capsule into the basket, raising the basket back into the cask, moving the cask back to the laydown area and sealing the cask, and loading the cask on a truck for shipment off-site.

Summary of Safety Evaluation: The concern in this evaluation is lifting heavy loads over the SFP and the possible consequences of dropping the load and damaging fuel or causing a leak in the SFP. This had been a concern that was raised by the NRC in Generic Letter 81-07. In order to address this issue, PBNP committed to implement the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This resulted in modification of the PAB crane to make it single-failure proof and further required additional safety factors and inspections of all lifting equipment used in the vicinity of the SFP. License Amendment No. 96 and 100 to Facility Operating License Nos. DPR-24 and DPR-27 for PBNP Units 1 and 2 also resulted and were implemented by a change to Technical Specification 15.4.14, "Surveillance of Auxiliary Building Crane Lifting Devices," and Technical Specification 15.3.8, "Refueling." This also resulted in a change to the FSAR which deleted Appendix F, "Consideration of Postulated Spent Fuel Cask Drop Accident," because it was no longer applicable. The requirements of NUREG-0612 and Technical Specification 15.4.14 are addressed in SMP 1066; therefore, the concern of dropping heavy loads in the SFP is not an issue. (SER 91-003)

116. SMP 1082, (Minor), Diesel Generator G02 Load Test, Revision 0, dated October 31, 1991. (New Procedure)

SMP 1082 will be performed during or following ORT 3, when plant conditions and equipment lineups are consistent with those required for the test. The SMP starts:

Diesel generator G02 in the fast start mode of operation transfers bus 2A06 from its normal offsite AC source (2A04) to G02. During the transfer, the loads on 2A06 will be deenergized for a short time. This is a normal occurrence during ORT 3; manually starts 2P-15B safety injection pump (in full-flow recirc mode), 2P-10B residual heat removal pump, 2P-38B auxiliary feedwater pump (in recirc mode), P-32D service water pump, P-32E service water pump, 2P-11B component cooling water pump, 2W1C1 containment recirc fan, 2W1D1 containment recirc fan; manually starts and stops safety injection pump 2P-15B with the diesel partially loaded; records diesel generator voltage, current, etc., 430 V bus voltage, and individual load current will be recorded during the test; and restores all equipment and systems involved in the test to their normal lineups.

Summary of Safety Evaluation: Some of the loads that will be started during SMP 1082, including the safety injection pump and containment fans, are not normally started during ORT 3 (The breakers are placed in TEST). Plant conditions allow the starting of the additional loads (SI pump is started in recirc) during the test.

During the performance of SMP 1082, equipment operability is maintained in accordance with Technical Specifications. All equipment and systems involved in the test are operated within their designed capabilities. Loading of diesel generator G02 is within its nameplate rating. The measurement equipment used during the test does not affect equipment operability. (SER 91-093)

117. ST-2, (Major), Core Cooling, Revision 2, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-041-01)

118. ST-3, (Major), Heat Sink, Revision 2, dated October 11, 1991. (Permanent)

Summary of Safety Evaluation: An administrative change places adverse containment setpoints before instead of after the normal containment setpoint. This is contrary to ERG usage and the deviation documents will address each specific case. By placing the adverse containment setpoint first, operators must decide that adverse containment conditions are not in effect before rejecting the first setpoint value and selecting the normal containment value. Since some EOP setpoints do not have an associated adverse containment value, previous experience has shown that operators incorrectly select the first value listed without addressing whether or not adverse containment conditions are in effect every time a setpoint is used. (SER 89-042-01)

119. ST-6, (Major), Inventory. A temporary change to Revision 0 lowers pressurizer level from 90% to 80% which results in entry into CSP-I.1, "Response to High Pressurizer Level," at this lower level of 80%. This is a more conservative approach instead of waiting until pressurizer level is 90%.

Summary o' Safety Evaluation: Technical Specification 15.2.3 requires the pressurizer level reactor trip setpoint to be ≤95% of span. The change from 90% to 80% still meets this specification. (SER 91-041-01)

120. ST-6, (Major), Inventory, Revision 1, dated June 11, 1991. (Permanent)

Summary of Safety Evaluation: The revision lowers pressurizer level from 90% to 80% which results in entry into CSP-I.1, "Response to High Pressurizer Level," at this lower level of 80%. This is a more conservative approach instead of waiting until pressurizer level is 90%.

Technical Specification 15.2.3 requ' es the pressurizer level reactor trip setpoint to be ≤95% of span. The change from 90% to 80% still meets this specification. (SER 91-041)

121. STPT 3.1. The setpoint for the turbine first stage pressure input to permissive circuit P-7, was changed to ensure that the TS turbine load setpoint for unblocking the "at power" trip is not violated due to instrument inaccuracies and drift.

Summary of Safety Evaluation: The necessity for changing the turbine first stage pressure P-7 setpoint is an outgrowth of the proposed increase in the TS required surveillance test interval for the reactor protection and safeguards and NIS instrument channels. During a historical review of the calibration results for the turbine first stage pressure P-7 related bistable functions, it was determined that the existing margin between the Technical Specification turbine load setpoint for P-7 and the setpoint may not be sufficient to accommodate an increased bistable drift between calibrations associated with an increased test interval. Therefore, in order to ensure that the Technical Specification setpoint is not violated due to an increased testing interval, the setpoint will be changed to provide a larger margin to the Technical Specification setpoint.

The turbine first stage pressure setpoint is changed from 9.0† % to 8.0† %. The new setpoint ensures that the P-7 permissive circuit functions as intended in the design objectives for the reactor protection system. (SER 89-146-01)

122. STPT 21.2, Unit 1&2, (Major), 480 V Breaker Overloads, Revision 4. (Permanent)

The amptector settings for the service water pumps were revised to provide additional margin for starting of the pumps under all conditions. Specifically, the instantaneous overcurrent setting was changed from 5.33 times the tap setting to 7.0 times the tap setting.

Summary of Safety Evaluation: The revised instantaneous pickup setting for the service water pump breakers reduces the probability that the breakers will trip on motor inrush current. Protection for short circuits and coordination with upstream overcurrent devices is maintained with the new setting. Justification and basis for the revised setpoint is documented in Calculation N-91-038. (SER 91-056)

123. TS-33, Unit 1, (Major), Surveillance Testing: Containment Accident Fan Cooler Units, Revision 7, dated November 12, 1990. (Temporary)

Summary of Safety Evaluation: The test satisfies the requirements of TS 15.4.5.I.C.2. This specification states that the containment fans shall be tested monthly to verify operability. The required service water flow through the coolers to verify operability is defined in FSAR Section 6.3.2 as 1000 gpm per fan cooler with three service water pumps operating.

Internal documents address the operation of the service water system on loss of offsite power with G02 failing to start. During this scenario, only two service water pumps would operate, the third SW pump on the G01 train would be out for service for maintenance. The service water header pressure would drop to 58 psig, which is less than the header pressures recorded in TS-33. TS-33 does not demonstrate the capability of the service water system to meet minimum design flow conditions under the scenario proposed. TS-33 will be revised to reflect these changes and performed to verify acceptability.

The fan coolers can be safely tested for this accident scenario provided all six service water pumps are available and only three are operated during the procedure. With six pumps available and a loss of offsite power, three service water pumps would operate even if a emergency diesel failed to start. (SER 91-024)

124. WMTP 11.59, (Major), Containment Fan Cooler Testing, Revision 0, dated June 28, 1991. (New Procedure)

Summary of Safety Evaluation: Internal documents address operation of the service water system upon loss of offsite power with G02 failing to start. During this scenario, only two service water pumps would operate, the third SW pump on the G01 train would be out of service for maintenance. The service water header pressure would drop to approximately 58 psig. The procedure limits the service water header pressure to approximately 60 psig with three service water pumps to simulate two service water pump operation and avoid pump vibration; balances flow through the fan coolers at >1000 gpm. Analysis of the test results will require correction for the pressure difference between test pressure (60 psi) and modeled pressure (58 psi); installs pressure gauges within the containment penetrations boundaries to record test pressure; and aligns service water overboard to only one unit to simulate actual plant operation.

The fan coolers can be safely tested for this accident scenario, provided all six service water pumps are available and only three are operated during the procedure. With six pumps available and a loss of offsite power, three service water pumps would operate even if an emergency diesel generator failed to start. (SER 91-024-01)

125. WMTP 11.60, Unit 2, (Major), Test Steam-Driven Auxiliary Feedwater Pump with Governor Sensing Line Disabled, Revision 0, dated November 9, 1991. (New Procedure)

WMTP 11.60 performs a fast start of 2P-29 turbine-driven auxiliary feedwater pump with the governor sensing line disabled to determine if the Terry turbine will overspeed. The test results will help determine if the governor sensing line can be permanently disabled alleviating the concerns raised in a JCO due to the governor sensing lines being high energy lines in the non-EQ AFP room.

Summary of Safety Evaluation: The governor sensing line does not have any significant affect on preventing a turbine overspeed. Although not anticipated, if the Terry turbine speed reaches ~4500 rpm, the mechanical overspeed trip device is actuated. This trip device has proven to be highly reliable during annual testing.

For personnel safety, all personnel in the AFP room will be instructed to stand clear of the 2P-29 cubicle during the initial fast start. Pump discharge pressure at the overspeed condition was found to be ~2200 psig. This pressure has been reviewed by the pump manufacturer and found to be acceptable for the limited time period it would exist. A review of the original piping code, B31.1-1967, also showed that this pressure is allowable for a limited time period.

The test does not affect the ability of the AFW system to deliver water to the steam generators. Only one AFW pump, 2P-29, is out of service during the test. The Technical Specification operability requirements pertaining to the AFW system are met at all times. The test is performed with steam generator pressures at approximately the no-load value of 1005 psig. 2P-29 is lined up for mini-recirc flow only. The governor sensing line is disabled by moving the mechanical travel stop on the Fisher regulator so it is not in contact with the governor valve's control linkage. The original position of the travel stop is recorded and verified. Upon test completion, it will be returned to its original position and this position verified. After performance of the test activities, a modified version of IT-09A is performed as a return to service test of 2P-29. (SER 91-101)

126. WMTP 11.61, Unit 1, (Major), Test Steam-Driven Auxiliary Feedwater Pump with Governor Sensing Line Disabled, Revision 0, dated November 27, 1991. (New Procedure)

WMTP 11.61 performs a fast start of 1P-29 turbine-driven auxiliary feedwater pump with the governor sensing line disabled to determine if the Terry turbine will overspeed. Terry turbine RPM will be recorded during the fast start. The test results will help determine if the governor sensing line can be permanently disabled alleviating the concerns raised in a JCO written due to the governor sensing lines being high energy lines in the non-EQ auxiliary feedwater pump room.

Summary of Safety Evaluation: According to the Terry turbine manufacturer, the governor sensing line does not have any significant effect on preventing a turbine overspeed. Although not anticipated, if the Terry turbine speed reaches ~4500 rpm, the mechanical overspeed trip device would be actuated. This trip device has proven to be highly reliable during annual testing. For personnel safety, all personnel in the AFP room will be instructed to stand clear of the 1P-29 cubicle during the initial fast start. Pump discharge pressure at the overspeed condition has been calculated and found to be ~2200 psig. This pressure was reviewed by the pump manufacturer and found to be acceptable for the limited time period it would exist. A review of the original piping code B31.1-1967 also showed that this pressure is allowable for a limited time period.

The test does not affect the ability of the auxiliary feedwater system to deliver water to the steam generators. Only one auxiliary feedwater pump, 1P-29, is out of service during the test. The Technical Specification operability requirements pertaining to the auxiliary feedwater system are met at all times. The test is performed with steam generator pressures at approximately the no-load value of 1005 psig. 1P-29 is lined up for mini-recirc flow only. The governor sensing line is disabled by moving the mechanical travel stop on the Fisher regulator so it is not in contact with the governor valve's control linkage. The original position of the travel stop is recorded and verified. Upon test completion, it will be returned to its original position and this position verified. After performance of the test activities, a modified version of IT-08A is performed as a return to service test of 1P-29. (SER 91-101-01)

127. WMTP 12.32, (Minor), Heat Retention Test of Fuel Oil Pumphouse, Revision 0, dated February 12, 1991. (New Procedure)

The test verifies the ability of the fuel oil pumphouse to remain above freezing during cold weather conditions if the pumphouse heaters are inoperable. It is performed by securing the pumphouse heaters and monitoring by thermocouple and recording the heat retention in the fuel oil pumphouse over a 48-hour period.

Summary of Safety Evaluation: Because the test is designed to be stopped before reaching 32°F, the fire protection system and fuel oil system are unaffected. By ensuring periodic inspection and recording, the test is secured after 48-hours or before falling below 34°F. (SER 91-005)

DESIGN CHANGES

1. MR 84-228*A, IWP 84-228*A-1 and IWP 84-228*A-2, DC Electrical Distribution. The purpose of this design package is to replace existing Westinghouse swing inverters DY0A and DY0B with new inverters manufactured by Solidstate Controls, Inc. (SCI).

Summary of Safety Evaluation: The new inverters are QA and were seismically and environmentally qualified (mild environment) by Wyle Laboratories. The new inverters have the same rating (10 kVA) as the existing inverters. The new inverters will be seismically mounted in the same locations as the existing inverters. The existing foundations will be modified to accommodate the new inverters. The new inverters have integral static transfer switches which can transfer the instrument buses to an alternate source upon inverter failure. Installation of the alternate source will be addressed by a different design package. The static transfer switches will not be operational until the alternate source is installed. The existing swing inverters supply lighting loads when they are not supplying instrument buses because they cannot be run at zero load. A transfer switch is used to supply lighting from either the inverter or a lighting panel. The new inverters can be run at zero load. Therefore, the inverter supply to lighting circuits will be removed. The lighting circuits will be supplied from the lighting panel breakers currently feeding the transfer switch. The transfer switch will be removed. During rewiring of the lighting circuits, temporary lighting will be provided. During the replacement of inverters DY0A and DY0B, the red and blue instrument buses will be fed from the normal inverters (1DY01, 1DY02, 2DY01, 2DY02) and will, therefore, not be affected by the installation. The new inverters have been seismically and environmentally qualified and will be seismically mounted. The new inverters will be mounted in the existing locations, maintaining safety train separation. The new inverters have the same ratings and will function the same as the existing inverters. (SER 91-083)

2. MR 85-252 (Unit 1), Auxiliary Feedwater System. MR 85-252 changes the annunciation provided by control switch logic for AFW pump P38A discharge MOVs. At present, if these control switches are placed in an intermediate position between full pushed-in and full pulled-out the automatic operation of the discharge MOVs is disabled without operators being alerted by annunciation of the "AFWS DISABLED" alarms. This modification changes the control switch logic so the alarm will annunciate as soon as the switch is removed from the full-in auto position.

MR 85-252*A eliminates the spring return left-to-center auto feature on the control switches for steam supply MOVs to turbine-driven AFW pump P29. MR 85-252 provides for annunciation of the "Unit 1 AFWS Disabled" alarm whenever 1P29 steam supply MOV control switches are placed in the maintained close position.

Summary of Safety Evaluation: To provide annunciation for discharge valve AF-4022 and AF-4023 control switches, the present auto-open switch contact will be removed from the AFW discharge MOV circuit. This contact then feeds an intermediate relay located in the P38A control circuit. Two sets of contacts off the intermediate relay replaces the auto-open switch contact and provide alarm annunciation as soon as the control switch is taken out of the full pushed-in auto position.

To provide annunciation for steam supply valve 1MS-2019 and 1MS-2020 control switches, the 1MOV-2019/2020 low suction pressure circuit is reconfigured so a set of switch contacts will feed an added alarm relay. Another set of contacts off the alarm relay provides alarm annunciation when the control switch is in the maintained close position.

The installation of additional relays does not substantially degrade reliability of the modified circuits. The added relays are of a type presently in use in QA applications. These relays have a good operating history and are Class 1E qualified. Mounting of all components will be accomplished similar to other comparable components in order to maintain seismic qualification of the control boards. This modification does not introduce a single failure fault since separation of train-related components and wiring is maintained. No other changes to the AFWS are made other than the above-described changes in control circuitry within the main control boards.

Operability testing of valves, control switches and annunciation will be conducted as part of the IWP controlling installation of circuit modifications. (SER 90-083A)

3. MR 85-260 (Common), PAB Crane. The modification provides the PAB crane with a radio control system.

Summary of Safety Evaluation: Addition of a radio control unit for the PAB crane will be designed to meet the single failure-proof requirements of NUREG-0612, Section 5.1.6 and Appendix C. The single failure-proof design of the Ederer X-Sam crane as described in the Ederer Topical Report EDR-1 (Revision 3) was found acceptable by the NRC in its August 3, 1983, safety evaluation; and the PBNP-specific design was accepted by the NRC in Amendment Nos. 96 and 100 (Units 1 and 2, respectively).

The radio control unit is designed so it will not bypass any single failure-proof circuits. As with the pendant, the radio control unit is designed so any improperly actuated controls will set the pneumatically-operated brake. Therefore, the single failure-proof design will be maintained. (SER 86-041)

4. MR 86-056*C (Common), Open Circuit Self-Contained Breathing Apparatus. This modification changes the air ports in C01 and C02 from being directly-supplied by service air to being supplied by dedicated air tanks in the north service building. The requirement is to provide four hours of breathing air to six control room personnel. This is based on fire protection considerations established by WE Industrial Health and Safety.

Summary of Safety Evaluation: SER 86-033 evaluated the change of breathing apparatus from Bio-Paks to Scott Air Packs. In addition, it evaluated the manifold that would supply dedicated air to the control room. As a continuation to that evaluation, the following details are evaluated.

MR 86-056*C removes the direct tie-in to service air and adds the independent emergency breathing air to control. The service air tie-in is available (via manual isolation) as a potential backup to the emergency breathing air. The manifold is used with the Scott air packs which have been previously discussed in the original evaluation. Two lines are routed to the control room from the north service building. These lines are separated in the turbine building to assure that they cannot be affected by the same fire. Separation is not applied in the cable spreading room or auxiliary feedwater pump room as fire protection is available in these areas. (SER 86-033-01)

5. MR 87-028 (Common), Service Water System. The modification installs isolation valves in the service water inlet line to the control room air conditioning condensers in order to accommodate single train operation during inspection or maintenance activities. In addition, the modification repipes the control room air conditioner condenser(s) Zurn strainer backwash line. The second proposed activity had been previously approved as modification request M-757 and was included within the scope of this modification.

Summary of Safety Evaluation: The control room and cable spreading room HVAC systems were not originally designed to be safeguards systems based upon the fact that if power is lost to the control room HVAC system, the equipment would operate for some time without cooling. However, in view of a NRC Information Notice (IN 85-79, "Potential Loss of Nuclear Instrumentation Following Failure of Control Room Cooling"), single-train AC condenser operation capability is recommended.

In order to perform this modification, service water to the AC condensers is secured for ~16 hours. During this time, temperatures in the control room racks and cable spreading room must be kept within the design range of the nuclear instrumentation system to ensure appropriate equipment operation.

Temperatures in these racks/areas are monitored while the air conditioning is out of service. Temporary forced ventilation will be available and used if necessary. This modification will be done during cool weather so the required area cooling loads are minimal. It is also possible to increase outside air intake and maintain the ventilating system operational to help supply cooling to the rooms. Both AC units will not be out of service at the same time so additional cooling is available between the two rooms via the control room/cable spreading room door, if necessary.

Taking the AC condensers out of service will have no effect on the control room/cable spreading room ventilation system other than cooling nor will it affect the emergency ventilation system (fans, dampers, etc. will be operable).

The modification is designed in accordance with B31.1 and supported to meet Seismic Class 1 design criteria. Materials used corresponds to those specified in the piping class specifications. Work will be controlled via special maintenance procedures. (SER 87-023)

6. MR 87-034 (Common), 480 V Electrical System. This modification replaces the existing 480 V DB breaker electro-mechanical overload protection with Westinghouse amptector 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 power for the new overload protection is provided by the current transformer type sensors, thus making the new protection independent of any outside power source. Since the new protection is solid-state versus the existing which is electro-mechanical, there is significantly less probability of system malfunction or erratic operation.

The solid-state amptector has the following characteristics: (1) There are no constantly moving parts to abrade and it is insensitive to vibration. (2) It is impervious to dirt, corrosion and humidity. All solid-state components have anti-corrosive protection and the tripping device is sealed against the entry of dirt. (3) Pickup is controlled by solid-state devices and will not change more than ±2%. (4) The device is temperature compensated from -20 to +90°C.

All units on safety-related buses (B03/B04) are qualified for seismic installation per IEEE 344. Those units installed on nonsafety-related buses (B01/B02), although not meeting the requirements of the IEEE standard, do meet the seismic criteria for PBNP as defined by a SQOG report. All installations will be per the Westinghouse instructions and addenda for retrofitted DB breakers. This will also assure that problems experienced with the direct trip actuator as described by NRC IE Information Notice 88-054 are corrected. Also, at least for safety-related breakers, full current testing will be performed after the modification to ensure all settings are correct and the new overload protection functions as expected. (SER 88-137)

7. MR 87-039 (Unit 1) and 87-040 (Unit 2): Waste Liquid System. The modification extends the sump "A" isolation valve test connection through the RHR cubicle and into the El. -19' general area. The change alleviates the personnel safety and ALARA concerns encountered when using the present connection for testing or sampling. At present, access to the sump "A" IVTC requires entering a high radiation area and standing on pipe ~7' above the floor.

Summary of Safety Evaluation: The addition is an extension of the containment boundary because it is upstream of the second containment isolation valves (two barriers are required for containment isolation). Tubing, Swagelok fittings, and Whitey valves will be used for this addition. The tubing and fittings will meet or exceed the pressure and temperature ratings of the system and the Whitey valve will provide sufficient shutoff capability. The Whitey valve will be installed under the seat, thus not relying on the packing for a seal. The tubing is run in a safe, relatively untrafficked area and therefore, is not likely to be inadvertently damaged. Redundant isolation for this line will be available by using the existing isolation valve test connection 1VCT-1409.

This addition increases the dead leg volume between the two containment isolation valves by approximately 20%, but that volume will be included in the Appendix J test volume and thereby be monitored during the Operations refueling test.

The increased volume will not pose a serious concern from either an operability, access or safety standpoint.

The addition is designed per B31.1 and is supported to comply with Seismic Class 1 criteria of the system. The addition will be leak-checked to ensure leak-tightness of the system and administratively controlled to ensure appropriate containment valve alignment. (SER 87-030)

MP 87-039 (Unit 1) and 87-040 (Unit 2): Waste Liquid System. The MRs extend the IVTC lines from the RHR pipeway out into the El. -19' general area. These extensions are terminated above a small sink for purposes of taking samples. The

original proposal simply collected the drainage from the sink in a bucket. However, due to the potential for contaminated liquids being present, it was requested that a closed drain line be routed from the sink to the El. -19' sump. This sump is a closed sump and the PAB ventilation system keeps a slight vacuum on it to remove any unwanted gases. A loop seal is incorporated in the drain line to prevent the release of gases back through the drain line, should the vent system fail.

Summary of Safety Evaluation: The sink and drain line is completely separate from the IVTCs. They will in no way affect the safe operation or seismicity of the IVTCs or any other piping/equipment in the area (by inspection). The sink and drain cannot adversely affect any other equipment should they fail, and therefore, they do not need to be seismically supported.

All materials that will potentially be in contact with radioactive liquids/borated water will be stainless steel.

This additional tap into the El. -19' sump does not present any additional flooding concerns in view of the existing floor drain. The RHR pump central area is drained to the El. -19' sump without any isolation device. This design feature will remain the same. (SER 87-030-01)

8. MR 87-121*F, Appendix R 480V Distribution System. This portion of MR 87-121 ties in the equipment installed under MR 87-121 design packages B, E, I, J, K, Q, and R with the existing plant equipment. This safety evaluation covers the adequacy and implementation of the sequential equipment tie ins and energizations as described in IWPs 87-121*F1 through *F15.

Summary of Safety Evaluation: IWPs 87-121*F5 and 87-121*F6, which tie-in the normal supply to Unit 1 residual heat removal (RHR) pumps, were reviewed and the concerns addressed in this SER. The other IWPs associated with this modification also address the concerns presented in this SER. This ensures an unreviewed safety question does not exist.

SER 87-054 covered the conceptual design of this modification. NRC SERs dated July 27, 1988, and January 11, 1989, concluded that the design of the alternate shutdown switchgear room conforms with the requirements of 10 CFR 50 Appendix R and the guidelines in Generic Letter 85-12. The design packages were reviewed for 10 CFR 50.59 applicability, and where required, a safety evaluation was performed to evaluate the impact of the equipment installation on the safe operation of the plant. SERs 90-116, 116-1, 116-2, 116-3, 116-4, and 116-5 were prepared for design packages E, Q, S, J, E and I, respectively.

Calculations N-89-035, N-89-036, N-90-044, N-91-043, N-91-044, and N-91-046 demonstrate the adequacy of the cables and the equipment to power and control the safety-related equipment.

The cable/equipment energized added per this design package are procured and installed to comply with the existing design criteria at PBNP.

When completely tied-in, this modification will not affect safety-related equipment or the operation of the plant. Adequate separation of normal/alternate sources are provided by seismically and environmentally qualified Class 1E switching devices.

Equipment which will be sequentially taken out of service for tie-in, is covered by TS requirements for redundancy, operability and limiting conditions for operations (LCOs). Should a failure (related to the hookup) occur during the tie-in period of the

safety related piece of equipment, the worst case result would be that the piece of equipment being tied-in would remain out of service. If under an LCO, the Technical Specifications would dictate what action to take if the equipment would remain out of service longer than the time allowed by the LCO. This postulated failure would not affect other plant systems.

For these same reasons, if an accident or malfunction of equipment important to safety previously evaluated on the FSAR occurred, the potential radiological release would not be increased.

For the safety-related plant equipment to be taken out of service, the LCO conditions, when applicable, and TS limitations are identified in IWPs 87-121*F1 through *F15. During the equipment tie-in period, the normal power feed to the load being affected will be re-established utilizing the affected switching device. This allows the exiting of an LCO (if applicable) and restoring the normal safety-related power source as soon as practical. The alternate supply is then tied-in under a separate LCO, if applicable, to test the alternate supply.

The 13.8 kV transformer and 480 V alternate shutdown bus energizations (IWPs 87-121*F1 and 87-121*F2) tie the alternate shutdown system to the 13.8 kV bus which supplies offsite power to the plants existing safety related buses. The alternate shutdown transformer and circuit breaker protection will be construction tested and verified prior to energization. The 13.8 kV bus will be split and redundant relaying would have to fail to result in a loss of offsite power to a single unit. The bus configuration and any required operator actions must be identified in IWP 87-121*F1. Thus, there is no significant increase in the probability of a loss of offsite power occurring.

The alternate shutdown switchgear and support equipment is not safety-related with the exception of the portion of safety-related cable being replaced in the safety-related normal supply and the Class 1E switching devices which tie in the alternate power source to safety-related equipment. The alternate power source cannot be used as a safety-related piece of equipment.

IWPs 87-121*F5 and 87-121*F6 were reviewed and are within the scope of this SER. The remaining IWPs are required to address the guidance provided in this SER. Thus, they remain outside the scope of this SER. With this omission, the sequential tie-in of the alternate shutdown power source to the plant equipment covered by this SER will not result in an unreviewed safety question. (SER 90-117-05)

MR 87-121*S, Emergency Diesel Generators. MR 87-121*S allows for the relocation of pressure switch PS-3057B and its associated tubing due to interferences with the designed mounting of a transfer switch added by MR 87-121*I.

Summary of Safety Evaluation: The pressure switch is associated with the G01 emergency diesel generator starting air system. This activity requires the starting air tank, T60A, and compressor, K5A, to be taken out of service. To minimize the impact of this change upon operations, this activity shall be performed while diesel generator G-01 is out of service for its annual maintenance outage.

The additional loading to the structures resulting from this modification was analyzed and determined to be acceptable. Pressure switch PS-3057B and its associated tubing maintain their seismic adequacy. (SER 90-116-02)

MR 87-121*T, Emergency Lighting Equipment. MR 87-121*T moves emergency light No. 10, located in the G01 emergency diesel room, and emergency light No. 77, located in the Unit 2 charging pump area near MCC 2B32, to locations which will not interfere with proposed conduit runs for the alternate shutdown system.

Summary of Safety Evaluation: In the G01 emergency diesel room, the lights will be seismically mounted to the wall via a P1001 cantilever support in approximately the same location as the existing light was mounted. The battery pack will be seismically mounted in a remote location near the lights. Emergency light No. 77 will be seismically mounted approximately 19" to the right of its present location.

Once reinstalled, the emergency lights will be tested. The supply breakers for the lights in the area will be opened, the emergency lights will be verified functional and redirected if necessary.

This modification is QA-scope. The emergency lights are presently seismically mounted in safe shutdown areas. The lights and the remote battery pack will be seismically mounted per design and installation guideline DG-E02. (SER 90-116-07)

9 MR 87-142 (Common), Cryogenic System. The modification adds instrument isolation valves and calibration tees to the cryogenic system to enable the calibration of pressure instrumentation needed to properly set and monitor the operating pressures in the system; specifically for PI-1A, PSA-1A, PI-1B, PSH-1B, PSL-1, PT-2A, and PT-2B.

Summary of Safety Evaluation: The FSAR states that for radioactive gas, "piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance."

In accordance with original design standards of the cryogenic system, stated in Instrument and Piping Standards, Nuclear Division, Class IC-N8, Swagelok fittings are used at all connections, except at the union of the tube lengths if necessary, which use socket weld fittings.

The FSAR statement on piping connections does not strictly apply in this case because the installation deals with tubing and not piping; but the installation still complies with the FSAR statement, as the Swagelok fittings facilitate valve and instrument maintenance. Bellows seal-type isolation valves are used as delineated in the FSAR for radioactive gas systems. The type currently in use on the valve panel per original design or equivalent will be used.

The instrument tees will be plugged with Swagelok plugs. All materials in contact with the gas will be in accordance with original design standards. The installation will be pressure tested to ensure leak-tightness before returning the system to operation.

Mounting details for the installation are Seismic Class 1 as described in Appendix A of the FSAR. Radioactive gas is controlled and necessary installation precautions taken through the use of a special maintenance procedure. (SER 87-049)

10. MR 87-168 (Common), Fire Protection System. MR 87-168 replaces the existing non-approved fire protection system for F-16, control room clean-up filter, with a system that conforms to NFPA standards. The existing system utilizes a non-listed ball valve, a weight suspended from the valve handle, and a wire with a fusible link suspending the weight, via the valve handle, to the plenum ceiling. The water is

sprayed through a spray grid using non-listed nozzles. The arrangement is supplied by service water through small bore piping that may or may not be seismic.

Summary of Safety Evaluation: According to FSAR 9.6.1 fire protection, the original fire protection system design was based on the guidance provided in the existing National Fire Protection Assoc ation. Additionally, the fire protection system is based on 10 CFR 50.48, 10 CFR 50 Appendix R, and on the recommendations of Nuclear Mutual Limited. Fire protection will be by an automatic fire suppression sprinkler head served by the plant fire water system instead of a manually-operated knife valve on the service water to the old spray grid.

The installation consists of a standard UL-listed, 180°F discharge sprinkler nozzle with a 175°F element. The nozzle will be supplied by fire water supplied from an existing header on the north side of the control building. The piping installed inside the control building will be installed seismically.

The installation is less susceptible to inadvertent actuation. The nozzle is not specifically designed as being seismic. However, the rugged construction and low mass make the probability of failure low.

Actuation of the nozzle will soak the carbon filter with water. This reduces the absorption capability of the filter. The filter will probably be declared out of service until a lab test can be completed on the charcoal. Technical Specification 15.3.12 is applied in this situation. This would be no different than an actuation of the existing spray system. The probability of the new system being inadvertently actuated is less than with the existing installation.

Since the rest of the plant conforms to NML standards and the fact plant fire protection has no problem with this change, the new valve configuration should be installed in accordance with NML Appendix 1.A.19, Section C, Suppression (1).

In regards to operator indication of a fire in the control room HVAC room, there is a heat-actuated detector mounted inside the charcoal filter cubicle (heat-actuated detector XS5174).

This change does not impact Appendix R compliance or commitments.
(SER 88-139-01)

11. MR 87-193*B (Unit 2), Rod Drop Stepping Test System. The MR installs a digital rod drop and stepping test system by permanently connecting a high speed multiplexer to the test points of the Unit 2 control rod drive stationary, lift, and moveable coil power circuits and the RPI coil circuits.

Summary of Safety Evaluation: This system replaces the visicorder but does not disallow its use. It will be comprised of 2 high speed multiplexers. One MUX is permanently connected to the back of the same test points in the rod drive circuitry that are used for rod stepping testing. The other MUX will be permanently connected to the same electrical node as the test points in the RPI circuitry used for rod drop testing via existing interposing terminal strip points. The input impedance (and DC resistance) is high enough so the existing circuitry will not be affected by the connection of these MUXs. The multiplexed signals will be transmitted to a personal computer which will generate plots of the signals from which rod drop time can be determined. The system also allows verification of proper rod stepping and rod full out verification.

Rod drop testing is performed each refueling outage to ensure that rods will drop within the 2.2 second time requirement as specified in TS 15.3.10.E and Table 15.4.1-2. Rod stepping testing is performed to ensure the rods properly step in accordance with TS 15.3.10.C.1.b. While these tests are required by Technical Specifications, there are no special QA requirements placed on the test equipment. The test equipment does not perform a safety function.

A short circuit fault on a rod position indication input signal would cause: a loss of the effected RPI indication, the associated rod bottom indicator to illuminate, a rod bottom rod drop annunciator, a plant computer deviation monitor alarm, a turbine runback, and an outward motion rod block. FSAR Section 7.3-7 discusses such a malfunction and states: "The digital and analog systems are separate systems; each serves as a backup for the other. The reactor operator may compare the digital and analog readings upon receiving a rod deviation alarm. Therefore, a single failure in rod position indication does not in itself lead the operator to take erroneous action in the operation of the reactor." Upon receiving all these indicators, the operator may conclude that a rod drop had occurred. AOP-6A provides the guidance to retrieve a dropped rod and in following this procedure, the operator would find that the rod had not dropped. AOP-6B provides guidance in determining if the RPI system is malfunctioning.

A fault on the control rod drive input signal would have some effect on the control rod drive system. The control rod drive circuitry has a current regulator that determines the amount of current a group of coils requires by monitoring the voltage across resistors in series with each coil in the group. The regulator determines the amount of current to provide the group through an auctioneered circuit that selects the coil with the greatest current demand (greatest voltage drop) and adjusts the firing time for the SCRs so as to provide this amount of current. If the faulted input happened to be the coil requiring the greatest amount of current before the fault, the regulator would change the amount of current supplied to the group based on the next most demanding coil after the fault occurs since the voltage drop across the dropping resistor would decrease due to the fault. In most cases, this change in current would not effect the operability of the rod group. Under the worst case scenario, a dropped rod could result. FSAR Section 14.1.3 evaluates a dropped rod accident. There would be no change in current if the faulted coil were not the most current demanding before the fault occurred.

The only means by which such faults could occur is through an installation wiring error or personnel error during subsequent maintenance. The potential for these two human errors can be eliminated through proper installation, testing and maintenance. The system will be installed and tested while Unit 2 is shutdown for refueling. Any installation faults will be found and rectified while the RPI and Rod Control systems are not in operation. Corrective or preventive maintenance will be performed in accordance with approved plant procedures by qualified personnel.

The new testing system will be hardwired to the existing RPI and rod control circuits. The present practice of making temporary connections for rod drop and rod stepping testing will no longer be required. Therefore, the probability of a short circuit (and hence the probability of an accident previously evaluated and the probability of a malfunction of equipment important to safety) is actually decreased. (SER 91-069-01)

12. MR 87-217*A (Common), Main Control Boards. MR 87-217*A provides control room indication for SA-9&10. SA-9&10 control room indicators will be wired in parallel to the local SA-9&10 indicators in the evaporator control panel. This modification satisfies a control room design review commitment. The addition of

indicating lights (and consequential cutting of holes) in main control boards 1C03 and 2C03 will not adversely affect the seismic qualification of the control boards.

Summary of Safety Evaluation: 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&10 valves. SA-9&10 will fail safely in the shut position, providing isolation of the radwaste system. This modification will not affect the operation of SA-9&10. (SER 90-110)

13. MR 87-219*B (Unit 2), Main Control Boards. This modification installs limit switches on the Unit 2 pressurizer spray valves and indicating lamps on 2C04 for control room position indication. In addition, a solenoid valve is installed in the spray valves' air line and a selector switch is mounted on 2C04 to provide the control operator a means to cause a Unit 2 spray valve to fail shut.

Summary of Safety Evaluation: The spray valves are QA-Scope items but not environmentally qualified. To prevent a limit switch lock-up from not allowing the valves to fully shut or fully open, which is highly unlikely due to inherent limit switch design features as evidenced by past history, the switches will be QA-Scope.

The solenoid valves are fail open (i.e., pass IA to spray valves) and non-QA due to the non-QA nature of the instrument air system. To avoid inadvertent spray valve closure, the solenoid valves will fail open on loss of power. Power will be supplied by a non-safety related bus (2Y05) so as not to unnecessarily load the station batteries or diesel generators.

Going to "override shut" is similar to going to manual closed on the controller when the controller and the positioner works. Thus, the licensing basis has not been changed. No auto feature changes. There is no significant impact on seismic qualification of the spray valve in view of similar applications.

However, going to "override shut" on the spray valve solenoid valve causes the valve to shut but will not provide a manual closed signal to controller PC-431H or PC-431C. It is, therefore, possible that when the "override shut" signal is removed, a wound-up controller could provide a full open signal to a shut spray valve.

The only time the override should be used is when the manual control stations cannot shut a spray valve. This circumstance can only be detected after making such an attempt. Thus, the manual control station should be sending a valve full shut signal when the override is used. Administrative controls can be used to ensure bumpless transfer if the spray valve controller is not in manual shut when the override is changed from closed to auto. (SER 88-108)

14. MR 87-227 (Common), Electric Generator/EDG. MR 87-227 modifies the main generator base adjust control switch and the emergency diesel generator (EDG) voltage regulator and governor control switches so the "raise" function is on the right and the "lower" function is on the left. Also included are the minimum and maximum excitation lamps, which are located directly above the base adjust controls. Lamp wiring is exchanged so the minimum excitation lamp will be oriented to the left of the maximum excitation lamp. The modification resolves HED 57, control room design review recommendation.

The 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, the rewiring of the meters to match the drawings returns the circuits to original design.

Summary of Safety Evaluation: Control switch changes involve the exchange of [illegible] at each switch and the installation of new nameplates. Indication lamp changes involve exchanging one wire from each lamp and exchanging the lamp labels.

Independent verification is performed to v[illegible] leads as they are lif[illegible] and reterminated. Continuity checks are used to verify proper switch o[illegible]tion and the integrity of the reterminated leads.

There are no seismic or Appendix [illegible] requirements. There are no system functional changes. (SER 90-048)

15. MR 88-012*B (Common), Circulating Water System. This design package installs two penetrations in Nos. 1, 2 and 3 condenser waterboxes (one in each waterbox).

Summary of Safety Evaluation: Work will be done in accordance with B31.1-1967 and approved plant welding procedure WP-4. Each penetration on the outboard side will be closed off by an isolation valve and a threaded cap. This precludes any flooding concern or significant waterbox air inleakage that could cause waterbox level to drop. (SER 88-044)

16. MR 88-018*F (Unit 1) and 88-019*F (Unit 2), Rod Speed Meter Replacement. MR 88-018*F/019*F replace existing rod speed meters. The existing meter is a Westinghouse Model WVX-252. The new meter is also a Westinghouse (now made by Weschler) Model WVX-252. The modification does not affect the function of the indicator.

Summary of Safety Evaluation, Unit 1: The rod speed meter is being replaced because it is inaccurate, and does not follow recommended human engineering guidelines as addressed by the CRDR committee.

Following installation, the meter will be verified functional and within tolerance by calibrating it per ICP 5.18.

This modification is QA-Scope. The rod speed meter is seismically mounted in main control board 1C04. The meter seismic mounting will be maintained by this modification. (SER 91-027-01)

Summary of Safety Evaluation, Unit 2: The provisions of SERs 91-027, 91-027-01 and 91-040 apply in their entirety for this design package. These SERs addressed corresponding MR 88-018*F for Unit 1. The only difference is that the Unit 1 references should be identified as being for Unit 2. (SER 91-027-03)

17. MR 88-022*A (Common), Radiation Monitoring. The modification provides permanent power supplies for frisker stations and installs five shielded frisking booths in the primary auxiliary building. The frisking booths will be located near the spent fuel pit, in both units' El. 66' fan rooms, in the central area of the El. 44', and in the El. 8' fan room. A spare frame, without shielding, is provided for use during infrequent jobs, such as blowdown evaporator outages or steam generator replacement or sleeving projects.

Summary of Safety Evaluation: The location of the semi-portable frisking booths is such that safety-related or seismic category structures or equipment (as described

in FSAR Appendix A) is not affected; or else the booths will be permanently attached (not portable; and seismically installed. The booths do not affect the NRC IE Bulletin 80-01 masonry walls.

Other issues that are addressed in the design and installation of the frisking booths and stations include: effects upon ingress/egress and personnel evacuation routes are minimized, existing equipment accessibility are maintained to the maximum extent possible, and the lowest local background radiation levels are sought for frisker booth and station location.

Each temporary use of the spare frisking booth shall be controlled in accordance with administrative controls. Design controls will be addressed by the administrative controls to ensure that the specific application of the temporary booth is appropriate. <u>(SER 88-026)</u>

18. <u>MR 88-030 and Design Package A (Common)</u>, 120 V AC Lighting. The modification installs lighting fixtures in front of each unit's ASIP panel in the control room and modifies existing lighting fixtures as required to accommodate the additional installations.

<u>Summary of Safety Evaluation</u>: The currently listed value of emergency AC lighting power is 50 kw. Adding the proposed lighting fixtures will increase the present load by 80 watts. Adding the 80 watts to diesel loading, however, is not significant for two reasons. First, the proposed fixture additions are actually reinstallations of fixtures that were removed when the ASIP panels were installed. Second, 20 watt "Wattsaver" light bulbs were recently installed in ten control room emergency lighting fixtures. This reduced the emergency lighting load by 40 watts per fixture, or 400 watts total. Similarly, because of these two reasons, control room heat loading is not a concern.

Installation of the new fixtures should be similar to that of the existing fixtures. The seismic adequacy of all control room lighting fixtures will be reviewed in accordance with NRC Generic Letter 87-02. <u>(SER 88-053)</u>

19. <u>MR 88-046 (Unit 2)</u>, Plant Shielding. This is an addendum which discusses the use of portable shielding racks in the regenerative HX cubicle.

<u>Summary of Safety Evaluation</u>: The racks will have a paint coating qualified to withstand post-accident atmospheres. The racks will be secured to a nonsafety-related structure (stairway) during power operations. The stairway is assumed to be Seismic Class 2. Securing of the racks to the stairway will be done with chains or heavy straps, and will be administratively controlled. The small load of the empty racks secured to the stairway are not a significant concern. Therefore, storing the racks in this location will not affect/impact any safety systems.

The lead blankets will be kept in a closed storage container (gang box). The container will be made of stainless steel. Storing the lead in this container does not exceed the floor's live loading limit. The container will be anchored in place (using QA anchorage), to prevent it from affecting/impacting any safety systems.

The storage of the blankets inside containment is acceptable from a fire loading standpoint, per the fire protection engineer. The blankets are rated to 225°F for extended periods of time. Accident conditions would result in temperatures in excess of this, but only for short durations. The container will be located in an area where it would not be subject to high energy piping failures. The storage container will be covered to prevent/minimize direct contact between the lead blankets and

containment spray. It will also be vented so even a severe pressure spike may, at worst, cause only minimal buckling\damage. These factors ensure that the lead blankets (and all material associatnd with them) will be contained and remain at this location, and will in not affect the containment recirculation sump.

This equipment decreases the free volume of containment by 25-30 ft^3. This is insignificant compared to the ~1 million ft^3 of volume presently in containment. (SER 89-112-01)

20. MR 88-065 (Common), Security. This modification contains safeguards information.

21. MR 88-067 (Unit 1) and 88-068 (Unit 2), Safeguards System. An evaluation was done to determine the acceptability of the ORT 3 test panels.

Summary of Safety Evaluation: The panels were installed in the early 1970s to accommodate testing described in TS 15.4.6.A.2. The purpose of the testing is to ensure that the diesel generators will start and assume load as described in FSAR Section 8.2.

The panels monitor the diesel loading by monitoring the position of vital load breakers, along with other relays. Cables connect the test points to either auxiliary contacts or cell switches on the breaker to monitor its position. A recorder is connected to the panels during ORT 3 tests to record the information. The test panels do not introduce a hazard to the various breakers and relays they are connected to because the cables are only energized for a short time during ORT 3 testing, which is done while the respective unit is in cold shutdown. While the unit is at power, the cables are deenergized.

The testing circuitry does not interconnect with any accident detection or mitigation circuits and will, therefore, have no effect upon the potential for or consequences of an accident. The cables doe not substantially increase the fire loading in the cable spreading room or the vital switchgear room and, therefore, do not present a fire hazard. (SER 89-067)

22. MR 88-093 (Unit 1) and 88-094 (Unit 2), Chemical Injection System. The modifications make the final electrical connections and testing of the Unit 1 and Unit 2 hydrazine and morpholine chemical injection systems. The addition of hydrazine from these systems will begin after testing and acceptance is completed in accordance with the IWPs. The use of the morpholine portion of the systems will not begin until after a separate safety evaluation report addressing morpholine use is completed. (SERs 89-103 and 89-103-01 addressed the mechanical tie-ins for these systems.)

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

A malfunction of the hydrazine controller would be detected by the residual hydrazine, oxygen or pH analyzers. The abnormal levels would cause secondary sample panel alarms which would 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 present hydrazine addition is accomplished by manual control of the pumps, thus proving the acceptability of manual pump control.)

The feedwater flow signal will be obtained by connecting tc the existing I/I converter connected to the steam generator "A" feedwater flow loop which currently provides a signal to the hydrazine pumps in the existing hydrazine addition system. It can be noted that since the existing hydrazine pumps are only operated in manual, the removal of the old signal cable may be accomplished at any time. The I/I converter provides isolation to protect the feedwater flow signal loop. As an added precaution, during installation the "A" steam generator main feedwater control will be placed in manual for a short time while the old feedwater flow signal is disconnected and the new cable is terminated. The IWP has been written such that the amount of time which the feedwater control will be in manual is minimal.

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

The fluid handling components of the system are compatible with the chemical concentrations which will be used. The pressure and temperature ratings of these components meet the system design ratings. The system design and construction was in accordance with B31.1-1967 which governed the design and construction of the original hydrazine addition system.

In order to prevent contaminants from entering the condensate and feedwater systems, the following installation steps and design features are in place:
a) System fluid handling components will be thoroughly flushed with batch tank water before injection into the condensate system takes place; b) the batch tanks are equipped with gasketed covers to preclude debris from entering the tank; c) t: chemical injection pump suction lines connect to the side of the batch tanks instead of the bottom to allow any debris entering the batch tanks to settle out on the tank bottom instead of entering the pump suction line; and d) the chemical injection pump suction lines contain Y-strainers which will filter out any particles which do get into the suction lines.

Shutoff valves and tubing lines are connected to the suction line Y-strainers and the pump discharge lines to allow flushing of the Y-strainers or venting of the pumps when necessary. These tubing connections along with the injection pump relief valve discharge tubing lines are routed to a funnel at the side of the skid. The tank drain header is also routed to this funnel. The funnel is routed to a turbine hall sump drain. This path of discharge is slightly different than that used in the existing hydrazine addition system, however both routings end up going to the retention pond in which the chemical concentrations are diluted before proceeding into the lake. The WE Environmental Department was informed of the change in discharge path and found it to be acceptable and intends to make the necessary notifications.

Hydrazine spills are addressed in AOP-12A. At the skid sites, the floor sloping and nearby turbine hall sump drain allow the chemicals to be contained within the plant boundaries. A service water hose connection is available within ~20' of each skid and could be used to wash down any spill. A subsoil drain manway exists just east of the Unit 1 skid location. The floor is sloped such that a spill of chemicals would not naturally flow toward the manway. However, if a catastrophic tank failure occurred, a portion of the initial slug of liquid could flow to the manway. In order to guard against this unlikely event, the manway opening will be sealed just below the manway cover. The seal is designed such that it could be temporarily removed if access to the manway is ever required. For personnel safety, an

eyewash/shower station will be installed at each of the skid locations. If these eyewash/shower stations have not been installed prior to initial use of the systems for chemical injection, then portable eyewash/shower bottles will be placed at the skid locations. The CHES sheets for the chemicals will be available at the skids.

The systems do not present an unacceptable increase in fire loading. The highest concentrations of the chemicals which will be present at the skid locations are 35% hydrazine and 40% morpholine. In these concentrations these chemicals do not burn. Chemicals at these concentrations are not considered flammable and therefore do not need to be kept in fire proof storage cabinets. If hydrazine spills are wiped up with rags, the system operating procedure instructs the worker to rinse the rags thoroughly with water after their use to ensure that the rags do not present a fire hazard as they dry out.

The systems will be operated by Chemistry. Training on the operation of the systems will be completed before acceptance of the systems. A hydrazine system operating procedure or morpholine system operating procedure will be approved for use before the skid is operated to inject that chemical. (SER 89-103-02)

23. MR 88-098*A (Unit 2), Safety Injection/Containment Spray System. MR 88-098 designs and installs new full-flow test lines in the Unit 2 high head safety injection, containment spray and residual heat removal systems. Design Package A provides test lines for the SI and CS systems.

Summary of Safety Evaluation: The new test lines in the SI and CS systems, installed in response to NRC Bulletin 88-04 with refinements added by NRC Generic Letter 89-04, allow for an increased flow rate through the pumps during inservice testing. The flow rate is increased to protect the pump from the adverse effects of hydraulic instability at low flow rates. The test lines will be isolated from their respective systems during normal operation by one isolation valve and one blank flange in series. The response to NRC Bulletin 88-04 concluded that since the majority of pump operations associated with miniflow conditions is related to the inservice testing program, the installation of higher capacity test lines will significantly reduce cumulative pump operation with less than recommended flow rates. The inservice test flow rates are increased for the SI pumps to >225 gpm per the manufacturer's recommendation and for the CS pumps to a point on the pump operating curve which is closer to pump operation during containment spray operation. The modification also adds recirc flow measurement instrumentation to provide local flow indication for inservice testing as described in NRC Generic Letter 89-04. This modification conforms with the description of the SI system in FSAR Section 6.2. (SER 91-074-02)

MR 88-098*B (Unit 2), Residual Heat Removal System. MR 88-098 designs and installs new full flow test lines in the Unit 2 high head safety injection, containment spray and residual heat removal systems. Design Package B provides a test line for the RHR system.

Summary of Safety Evaluation: There is an existing section of pipe between the train "B" heat exchanger discharge piping and 2SI-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 regulating butterfly valve. The piping complex also ties into the train "A" heat exchanger discharge piping. The test line is installed in response to NRC Bulletin 88-04 with refinements added by NRC Generic Letter 89-04, allows for an increased flow rate through the pumps during inservice testing. The flow rate is increased to protect the pump from the adverse effects of hydraulic instability at low flow rates. The test line will

be isolated from each train of RHR during normal operation by the isolation valve and a blank flange, replacing the flow measuring orifice, in series. The response to NRC Bulletin 88-04 concluded that since the majority of pump operations associated with miniflow conditions is related to the inservice testing program, the installation of higher capacity test lines will significantly reduce cumulative pump operation with less than recommended flow rates. The inservice test flow rates are increased to >520 gpm per the manufacturer's recommendation. This modification conforms with the description of the RHR system in FSAR Section 6.2.
(SER 91-074-03)

MR 88-098*A&B (Unit 2), Safety Injection System. ECRs NE-91-334 and NE-91-335 propose changes to MR 88-098*A and B to move the NUREG-0578 boundary for the SI, CS and RHR systems from the newly installed blank flange to either the upstream isolation valves, 2SI-329A&B for SI, 2SI-862G&H for CS, and 2SI-706A&B for RHR, or the newly installed blank flange. MR 88-098*A and B have been evaluated under separate safety evaluations.

Summary of Safety Evaluation: The change clarifies and provides options for NUREG-0578 system boundaries required to be intact during post-LOCA coolant recirculation. The consequences of a LOCA are dependent upon the integrity of NUREG-0578 boundaries during post-LOCA coolant recirculation. The boundaries that are designated will be tested and the actual leakage will meet the established acceptance criteria stated in FSAR Section 6.2. If any blank flange is the designated boundary, there is the possibility during inservice testing of the systems that an accident could occur when NUREG-0578 boundary is not intact, since the orifice plate will be installed rather than the blank flange. In this case, the test procedure will direct the operators to secure the testing and shut the isolation valves upstream and downstream of the spectacle orifice/blank flange. This will provide a double isolation boundary to prevent highly radioactive fluid leakage back to the RWST. Since operator action outside the control room is required to place the SI, CS and RHR systems on recirculation, operator action can be relied upon to shut the test line isolation valves before the systems are operating in the recirculation mode.

The requirements of NUREG-0578, to minimize leakage from systems which will be recirculating post-accident highly radioactive water, are met with this change. System operation during normal and accident conditions is not affected.
(SER 91-074-07)

MR 88-098*E (Unit 2), Safety Injection System. The design package removes the lower rings of packing on 2SI-868A&B and replaces them with a stainless steel spacer. The upper rings of packing (above the lantern ring) serves as the stem packing. In addition, the existing 1/2" leakoff connection is configured to accept a plug or 1/2" test line.

Summary of Safety Evaluation: The valves are being modified to allow leakoff testing to be performed on them using the leakoff line as a pressurization point to pressurize the valve body between the disks. This is being performed to allow Appendix J testing to be performed in the proper direction. This is required to allow them to be used as containment isolations during flow testing of the containment spray system.

Replacement of the lower rings of packing with a spacer was previously evaluated. The evaluation concluded that the use of five or more upper rings of packing (upper being defined as above the lantern ring and leakoff connection) was sufficient to provide a packing pressure boundary around the stem. Also, the operability and

seismic capability of the valves will not be affected since no additional possibility for valve binding exists and no significant weight change will occur. The replacement will be controlled in accordance with the guidelines of maintenance procedure MI 32.2, which details the requirements for replacing the lower rings of packing in deep stuffing box valves with spacers. The only exception will be the use of a stainless steel spacer instead of a carbon spacer to alleviate a concern on carbon dust entering the containment spray system. (SER 91-074-06)

MR 88-098*F (Unit 2), Safety Injection System. The design package adds local reading flow transmitters for the SI, RHR and containment spray test lines. Flow transmitters are required for obtaining pump flow data during ASME Section XI testing. Installation of transmitters will occur when the RHR, SI and containment spray systems are out of service for installation of test line piping.

Summary of Safety Evaluation: The transmitters allow Operations to monitor test line flow rates for ASME Section XI inservice testing. Test lines and transmitter sensing lines will be isolated during normal plant operation. Since the transmitters and sensing line tubing is seismically mounted, reasonable assurance is provided against primary system leakage via the transmitter and sensing line if a LOCA and seismic event occurred while the test lines were in service. Transmitters and sensing line represent a closed system which will contain primary coolant should the isolation valves leak by.

The transmitters do not perform any safety-related function, and will be powered from local lighting panel receptacle circuits. All conduits will be installed as Seismic Class 1 to avoid potential impact with safety grade equipment located in the surrounding areas. The transmitters meet the accuracy and range requirements of Appendix C of the PBNP Inservice Testing Program (PPR-10). (SER 91-074-04)

24. MR 88-099 (Unit 1), Auxiliary Feedwater System. This modification adds flow measurement instrumentation to the recirc line and increases the capacity of the recirc line for the auxiliary feedwater pump 1P29. MR 88-099 was initiated in response to NRC Bulletin 88-04 with refinements added by NRC Generic Letter 89-04.

Summary of Safety Evaluation: The capacity of the recirc line is being increased from the present 30 gpm to approximately 116 gpm. The original recirc line capacity was established solely on the basis of pumped fluid temperature rise. In order to protect the pumps from the effects of hydraulic instability at low flow rates, the capacity of the recirc line will be increased per the recommendations of the manufacturer, Byron Jackson Products, to a minimum of 100 gpm. To meet the requirements of ASME Section XI testing, flow indication will be added on the recirc lines.

Calculation N-91-032 estimated the effect the increased size of the new recirc line would have on the flow rates to the steam generators if AF-4002 sticks open. 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 FSAR accident analyses is still available. In addition, the manual valve AF-15 can be used to isolate the recirc line or air to the control valve can be isolated. Calculation N-91-007 was used to verify that there is more than 5 minutes for manual action to isolate the recirc line. AF-4002 has position indication in the control room so this failure can be identified. AF-4002 was recently added to the ASME Section XI test program and is verified to open and close on IT-290.

The modification for 1P29 will be completed during the 1991, Unit 1 refueling outage. The isolation for this work will result in the recirc lines for pumps P38A, P38B, and 2P29 being out of service in addition to the 1P29 recirc line. P38A, P38B, and 2P29 pumps will not be out of service since discharge paths to the steam generators will be lined up to compensate for the isolation of the recirc lines. Administrative controls will be established to minimize the potential for damage to the P38A, P38B, and 2P29 pumps during the time that the tie-in to the common recirc discharge line is installed for 1P29. These administrative controls are presented in detail in the installation work procedure. Installation will be completed so as to minimize the time that the recirc lines for the P38A,P38B, and 2P29 pumps are out of service. (SER 91-025)

MR 88-099*B (Unit 1), Increased Auxiliary Feedwater Pump Mini-Recirc Line Flow Capacity. The design package replaces the existing mini-recirc lines of the auxiliary feedwater (AFW) pumps with new larger capacity min-recirc lines. The flow rate is increased to protect the pump from the adverse effects of hydraulic instability at low flow rates. The modification also adds recirc flow measurement instrumentation to provide local flow indication for in-service testing of the AFW pumps.

Summary of Safety Evaluation: The capacity of the mini-recirc lines for the motor-driven AFW pumps will be increased from 30 gpm to a minimum of 70 gpm based on the recommendation of the manufacturer. The increased flow through the mini-recirc line is to protect the pumps from the adverse effects of hydraulic instability at low flow rates. After each AFW pump is up to rated speed and flow, a shut signal is sent to its associated mini-recirc valve which shuts after a 3 minute delay time. Calculation N-91-069 shows that each motor-driven AFW pump will deliver 111 gpm with its mini-recirc valve full open vice the nominal 200 gpm flow. Calculation N-91-032 shows that each turbine-driven AFW pump will deliver 324 gpm with its mini-recirc valve full open vice the nominal 400 gpm. If we apply single failure criteria to the safety- related components of the AFW system, and the single failure is either AF-4007 or AF-4014 sticking open after the 3 minute delay time, each turbine-driven AFW pump at 324 gpm for 3 minutes and 400 gpm thereafter, one motor-driven AFW pump at 111 gpm for 3 minutes and 200 gpm thereafter and the other motor-driven AFW pump at 111 gpm provide adequate flow to the steam generators for decay heat removal.

For the loss of normal feedwater due to a seismic event, calculation N-91-007 shows that the increased size of the mini-recirc line does not degrade the decay heat removal capability during the event.

The new malfunction is within the scope of the analysis for the normal loss of feedwater accident. Both AF-4007 and AF-4014 are included in the ASME Section XI test program and are verified to operate properly by IT-290

This modification will be completed with both Unit 1 and Unit 2 at power. For a period of time during the installation, all four AFW mini-recirc lines will be isolated. During this time, all four AFW pumps will be considered inservice and operable because the minimum flow requirements for the AFW pumps will be ensured through discharge paths to the steam generators. For the steam-driven pumps, the valves in the discharge paths to the steam generators are normally open. For the motor-driven pumps, the normally closed diaphragm operated valves (AF-4012 and AF-4019) in discharge paths for their respected pumps will be throttled open to provide a discharge path to the steam generators. To complete the installation, one motor-driven AFW pump will be taken out of service and then returned to service one at a time under the provisions of TS 15.3.4.C. Since the TS allows for a

motor-driven AFW pump to be taken out of service for maintenance or testing for a period of 7 days, there is no reduction in the margin of safety. Prior to placing the AFW pump back in service the testing required by TS 15.4.8 will be completed.

The AFW system is a Seismic Class 1 system. NRC SER dated September 16, 1986, requires that the mini-recirc lines be Seismic Class 1 to the second isolation in series from the discharge of the AFW pump. The piping and supports for the completed modification have been designed to meet the Class 1 requirements. During the installation on the mini-recirc lines, temporary supports will be installed as needed to maintain the seismic qualification of any inservice AFW pump. In addition, due to the mini-recirc line modifications, two supports in the motor-driven AFW pump discharge lines (DB3-2H7 and DB3-H11) and one support in their cross connecting line (DB3-2H6) will require modification. Temporary supports will be installed, as needed prior to beginning modification work on any of these supports to maintain the seismic qualification of any inservice AFW pump. (SER 91-025-03)

25. MR 88-160*A (Unit 2), Containment Structure. Modification 88-106 installs a permanent penetration for both mechanical and electrical connections that are needed to support steam generator refueling maintenance and testing work. The design package installs a preliminary penetration into an existing spare penetration; El. 32', pipeway No. 3. The existing caps will be removed, both inside and outside of containment. The caps will be replaced with 150 lb ANSI B16.5 flanges. This penetration will contain the electrical cables and hoses needed for steam generator outage work.

Summary of Safety Evaluation: The revised configuration meets the design, installation and testing requirements of the original containment penetrations. This includes compliance with ASA N6.2-1965, "Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors," and NUREG-0737 containment isolation requirements (Section II.E.4.2). The penetration will be painted on the inside of containment.

During times when containment integrity is required, the flanges will be blanked off both inside and outside of containment. The two blank flanges will provide the redundant containment barriers. To verify leak-tightness, the assembly will be volumetrically tested. When containment integrity is not needed but fuel motion is in progress, the penetration will be sealed with foam. The foam will be designed to provide protection for a refueling accident by providing a sealed penetration for HVAC pressure. Mid-loop concerns can be resolved by disconnecting the hoses and cables on the outside of containment and bolting on the blank flange in the auxiliary building within the half-hour time frame of OP-4F. (SER 89-106)

26. MR 88-168 (Unit 1) and 88-169 (Unit 2), Electric Generator. The modifications add a redundant vapor extractor on the generator bearing drain tank. This redundancy more positively prevents the possible buildup of hydrogen gas in the drain tank or bearing pedestals since it would be available for operation if the running unit failed.

Summary of Safety Evaluation: The physical characteristics and performance requirements of the new vapor extractor closely match those of the existing extractor.

The new extractor will have a more reliable drive train design than the existing unit. The inlet and outlet of the new extractor will be tied into the existing system per B31.1-1967. The new extractor will share instrumentation and control board indicating lights with the existing extractor, which are needed for proper operation of the system. Power will be obtained from the same motor control center

(1[2]B41) that powers the existing extractor, but since only one extractor will be in service at a time, it will not change the overall load on the motor control center. The change will not affect the functionality of the system, but will provide added reliability and availability. (SER 90-020)

27. MR 88-170 (Common), Fuel Oil System. MR 88-170 adds a recirc flow path and the necessary instrumentation to the fuel oil transfer pumps so they can be tested in the IST program. It involves a suction pressure gauge, changing the 1" gravity feed line to 2", and adding a line which contains a flow gauge and runs in parallel to the gravity feed line. The parallel 2" lines are provided with a floor support.

Summary of Safety Evaluation: The modification is designed in accordance with B31.1-1967 and the original design specification for Class HB piping. A detailed seismic analysis of the fuel oil system piping was performed. As a result, a floor support is added for the modified piping at each pump. The added floor support assures that the pump nozzle loading is minimal. With the addition of this support, the as-modified configuration is better than the existing configuration from a piping and support stress standpoint. All materials used in the installation of the modification will be QA.

This modification does not introduce any additional discharge flow paths which could decrease flow to the EDG day tanks. The effective size of the gravity feed bypass line is increased because it is being replaced with two parallel two inch lines. This could be construed as an increase in the potential flow back to the emergency tank, instead of to the day tanks, in the event that one of the isolation valves would leak. These lines will be tested as leak-tight piping runs, similar to the IST concept on other safety related systems. (SER 90-062)

28. MR 88-171 (Unit 1), Instrument Air. The modification upgrades the instrument air supply to the crossover steam dump valves in accordance with NRC Generic Letter 88-14. The modifications involve 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. The accumulator volume will be sufficient to open the four dump valves.

Summary of Safety Evaluation: The modification ensures that 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 with 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 remains 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. This modification increases the reliability of this system by assuring that the support system to operate the valves is available. (SER 90-098)

29. MR 88-175 (Unit 1) and 88-176 (Unit 2), Containment Structure. The steam generator primary manway diaphragms are removed during each steam generator primary side inspection. These diaphragms are normally in contact with reactor coolant, and are highly contaminated. Typical dose rates for a diaphragm are

10 R/hr gamma contact, 100 R/hr beta contact, 1 R/hr gamma at 18", and 7 R/hr beta at 18".

MR 88-176*C provides lockable, shielded storage containers for the diaphragms on each containment 10' platform.

Summary of Safety Evaluation: Past practice has been to store the diaphragms behind a concrete shield wall inside containment, shield them with lead blankets, rope off and conspicuously post the area, and install a flashing light as a warning device. This practice, while it is allowed by plant procedures and Technical Specifications, is not desirable and possibly not legal for routine future applications (reference NRC IN 88-79).

The lockable, shielded storage containers for the steam generator primary manway shields meet the requirements of NRC IN 88-79. (SER 89-120)

30. MR 88-183 (Common), Fire Protection/Fire Water Systems. MR 88-183 covers one addition and one relocation of sprinklers in each diesel generator room. This modification is necessary due to the obstruction of sprinkler spray pattern by monorails and fan housings.

    Summary of Safety Evaluation: The hydraulic effects were evaluated and system capacity is not impaired. Room drainage requirements remain unchanged and the new sprinkler locations does not increase the effect of inadvertent system operation. One hanger was added to the support system. Standard supports were noted to provide adequate support of fire protection piping during seismic events. (SER 91-033)

31. MR 88-184 (Common), Fire Protection. The modification replaces the gas turbine building carbon dioxide fire suppression system with a dry pipe sprinkler system. Changing the type of fire suppression system in the gas turbine building changes the method of suppressing a fire in the building. The gas turbine and related equipment could be affected by a fire or possible inadvertent actuation of the suppression system. The gas turbine is a source of standby power and is not used for plant operation.

    Summary of Safety Evaluation: Open head sprinklers on the X04 transformer system will be replaced with closed head sprinklers. Heat detectors, which actuate the carbon dioxide system, will actuate deluge valve 3707 to allow water to enter the header. Heat from a fire will fuse the sprinkler leads to initiate fire suppression.

    The redundant component actuation necessary to initiate fire suppression provides adequate protection against inadvertent actuation and water poses less of a personnel safety hazard than carbon dioxide.

    FSAR Section 8.1.1 lists the gas turbine as a source of standby power and states that the gas turbine is not committed as a source of emergency power. The gas turbine starting diesel provides backup power for the TSC. NUREG-0696 requires a TSC power supply reliability of 99%, which is achieved with the normal power supply.

    X04 sprinklers are provided primarily for fire containment. The use of closed head versus open nozzles will not result in an appreciable delay in response time. Changeout prior to divider wall installation will not result in a reduced level of protection. (SER 89-091-01)

32. MR 88-188*B (Common), Miscellaneous Motor-Operated Valves. The modification involves three different motor-operated valve (MOV) betterment changes. The changes should take approximately four years to complete and design packages will be utilized to modify a limited number of MOVs during each outage. The three changes consist of: a) Installing four-rotor limit switches to allow for the complete separation of position indicators and torque switch bypasses. This will increase the accuracy of position indication light, while eliminating the occurrence of premature tripping of torque switches due to inertia during startup of the operator; b) installing T-drains to Limitorque operators to provide a reliable and qualified way to remove damaging foreign liquids from limit switch compartments; and c) providing overload indication to the control room.

Summary of Safety Evaluation: The operating time and characteristics for any and all of the valves will not change with the introduction of the new four-rotor limit switch. The change enhances MOV position indication and torque switch bypass operation only. The function and operation of interlocks remains the same. The four-rotor limit switch is a standard Limitorque-supplied component and is already used in many MOVs in our plant.

The T-drain is both EQ and QA for MOV operators. Existing weep hole drains are not qualified in some locations and they provide a direct path to the internals of the operators. These plugs need to be replaced. The parts, the T-drain and pipe plug will add no extra weight and are seismically qualified in the MOV. (SER 90-039)

MR 88-188*C (Unit 1), Miscellaneous Motor-Operated Valves. This design package installs T-drains on all the motor-operated valves installed in Unit 1 containment and on all Unit 1 safety-related valve operators installed outside of containment.

Summary of Safety Evaluation: Adding T-drains to safety-related or nonsafety-related valve operators provides additional protection from moisture intrusion. The addition of the T-drain does not adversely affect the EQ rating of an MOV. This will have no affect on valve or motor operator structure, qualification or function. (SER 90-093-01)

33. MR 89-023*C (Unit 1) and 89-024*C (Unit 2), Reactor Coolant System. The design packages replace the existing Tygon tubing used for local RCS reduced inventory level indication with a stainless steel magnetic level indicator. The magnetic level indicators are vented at the top as was the Tygon tubing to avoid creating a syphon and are used per procedure the same as the Tygon indicator was. The changeout is expected to improve the reliability of the local level indication due to the more durable, permanent construction involved with the magnetic level indicator as compared to the Tygon level indicator.

Summary of Safety Evaluation: The level indicator will be installed during a refueling outage when reduced inventory level indication is not needed. The new level indicator will be calibrated against actual water level as measured from the existing elevation reference marks used with the Tygon level indicator. The manufacturer's literature states an accuracy of ±0.02" for the level indicator.

All of the new wetted components are stainless steel. The level indicator has a pressure rating of 800 psig while all of the other new components have ratings of at least 3000 psig. Since the level indicator is always vented at the top and is only valved in when the RCS is depressurized, the maximum pressure which the new component could see would be from the water head of a full level indicator (<10 psig). Since the level indicator will be located inside containment near safety-related equipment, it will be supported seismically such that it will withstand the

hypothetical earthquake without catastrophic failure. The process tubing lines are also supported seismically.

Changeout c. the level indicator is expected to improve the reliability of the local indication. A leak in any of the new components could produce incorrect indication of the local level indicator and control room indications LI-447 and LI-447A. However, the probability of a leak is considered extremely low due to the use of components with pressure ratings well in excess of the maximum possible pressure that could be seen by the vented level indicator and due to the performance of an initial service leak test. The newly-installed components are expected to provide a more reliable pressure boundary than the Tygon tubing did. Even if a leak occurred it would give a conservative indication of a lower than actual level. In addition, it can be noted that backup/indirect indications of adequate core cooling (core exit thermocouples, RHR system parameters) are not impacted by the modification.

To prevent boric acid buildup on the inner surface of the level indicator which could affect the ability of the float to move freely up and down the standpipe, the level indicator will be flushed with DI water at the end of each outage. The test connection and drain valves installed in the process tubing lines will facilitate this flushing. It can be noted that if the level indicator was affected by boric acid buildup during use, it would be discovered by comparison with redundant control room indications LI-447 and LI-447A and flushing could be carried out at that time. (SER 89-054-02)

MR 89-023*C (Unit 1) and MR 89-024*C (Unit 2). RCS Level Indicator. This addendum to SER 89-054-02 documents the alternate flow path which will be used with 1(2)LI-447B. This flow path is from the intermediate leg of the reactor coolant system through valves RC-523, RC-524 and RC-522A. This flow path significantly improves the response time of the level indicator as compared to its response time if connected off the RVLIS variable leg. The alternate flow path also provides a check of reactor vessel level which is completely independent of instrumentation channels LT-447 and LT-447A.

Summary of Safety Evaluation: The local level indicator (LI-447B) is used as an independent check of the LT-447 and LT-447A instrumentation channels. When level is being changed only LI-447 and LI-447A indications are used. After level has stabilized, the local indication is checked. The alternate flow path will be isolated per procedure from the variable legs of LT-447 and LT-447A; therefore, the local level indicator provides a completely independent means of checking the electronic channels.

Seal injection to the "B" RCP for Unit 1 or "A" RCP for Unit 2 produces a false high reading by the level indicator. Therefore, the operating procedures will secure seal injection to the appropriate RCP prior to using the level indicator. In addition, the response time of the level indicator would be increased if the RCS was drained below 3/4 pipe due to the diffuser height in the RCP through which the RCS level would have to equalize. This will also be noted in the operating procedures.

With exception to the alternate flow path described above, the safety evaluation performed in SER 89-054-02 remains unchanged. (SER 89-054-03)

34. MR 89-033 (Common), EDG and Air Compressor Rooms. This modification adds some ladders and platforms to the heating units for the EDG rooms and the air compressor room. The additions are being made to upgrade personnel safety for access to the heating units.

Summary of Safety Evaluation: Installation of this modification will not increase the probability of an accident. The modification will have no impact on the reactor, the reactor coolant system, or any of its control systems. The only equipment impact that this modification will have is with the heating units.

The modification will not increase the consequences of an accident. A worst-case malfunction that this installation could cause is inoperability of the heating units. If any of the heating units were to fail, the plant would still be within its design basis because the heating units are not required to assure EDG operability. Therefore, the EDGs will still be available to mitigate the consequences of an accident as described in the FSAR. There are no other impacts that this installation will have on the consequences of an accident. (SER 90-007)

35. MR 89-034*B (Unit 1) and IWP 89-034*B, Radwaste Steam. The modification and IWP install new steam traps upstream of SA-9, 1MS-2019 and 1MS-2020. They also change out the 1" line which connects the discharge of ST-SA-21 and 22 to the Unit 1 LP trap header. The new line will be 1-1/2" stainless steel. The new steam traps will be routed to this new line. The modification also replaces the 1" line which currently drains the radwaste steam traps (ST-21 and 22) to the Unit 1 LP trap header. This is necessary because the new steam traps will be routed to this line and it will then be undersized. It is also necessary to upgrade the design pressure of this line to make it consistent with the rest of the Unit 1 LP trap header (1085 psig). It will be replaced with stainless steel for erosion/corrosion concerns. Finally, one of the two valves which currently tie these drains to the P-90 condensate return unit will be removed and capped.

Summary of Safety Evaluation: Installation of the new steam traps provides better trapping capability for the radwaste steam system and for the AFP steam supply line and governor sensing line. All of the lines are currently trapped by being routed to the governor sensing line, which has one steam trap at the bottom. The MR provides individual steam traps which are sufficiently sized for the anticipated operational and warm-up condensate loads. By removing these condensate loads from the governor sensing line, the reliability of the AFP is increased because the steam lines will not be at such high risk of being filled with condensate if they are called on to deliver steam to the AFP.

The modifications are being installed per the original design and subsequent enhancements of the affected systems. The steam inlet will be designed to 1085 psig with carbon steel materials. The steam trap outlet will also be 1085 psig but will use stainless steel. Use of stainless steel at the discharges will alleviate most of the concern with erosion/corrosion.

The steam trap assemblies being installed upstream of the MOV-2019/2020 are being installed such that they do not affect the seismic rating of the current system. The first valve on the 3/4" line from the 3" AFP steam line is designed to be the seismic, QA, and safety-related boundary. This is done so the installation will have 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 occurred, its consequences would be 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 BASTs and MOV-2019/2020. The EQ analysis of this area already assumes a steam environment due to a line break. Addition of the 3/4" steam trap lines will 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 would be affected by a break in these lines. Therefore, EQ and HELB are not a concern. (SER 90-087-02)

MR 89-034*D (Unit 2), Main Steam System. ECR PB-91-088 adds an isolation valve in the 2P-29 governor sensing line. This valve is added to facilitate a possible solution to a JCO written due to the governor sensing lines being high energy lines in the non-EQ AFP room. It is planned to test 2P-29 after U2R17 to see if the governor sensing line is required to prevent the Terry turbine from overspeeding. If the line is not required then this valve may be used to isolate the line alleviating the JCO concerns with the 2P-29 governor sensing line.

Summary of Safety Evaluation: The new valve (2MS-00245) will be installed in the 2P-29 3/4" governor sensing line on PAB El. 46'. This valve will be added to the system checklist and will initially be locked open and independently verified. Any subsequent positionings of this valve will be covered under separate procedures or under MR 91-220. The valve allows the 2P-29 governor sensing line to be isolated separately from the steam traps for 2MS-2019 and 2020. (SER 90-087-03)

36. MR 89-056*A (Common), Lifting Devices. The modification provides a generic lifting pad eye for lifting plant equipment such as valve operators and internals. Although this equipment is not specifically addressed in the PBNP FSAR, an evaluation was performed as documented in SER 90-032, which was approved in MSSM 90-07. This addendum involves MR 89-056*A, which addresses the design of two new pad eyes that are for increased loads.

Summary of Safety Evaluation: The lifting pad eyes will be used to facilitate maintenance practices that were considered as part of the design of the plant. The design of the lifting pad eyes is based on MI 7.1 and design guideline DG-C-01. The original safety evaluation addressed a lifting pad eye designed to support a 1000 lb load. This addendum addresses a Type A lifting pad eye designed for a 1000 lb load; a Type B lifting pad eye designed for a 5000 lb load; and a Type C lifting pad eye designed for a 3000 lb load. A specific seismic evaluation was not done because the system must be out of service when the lifting pad eye is being used. When the lifting pad eye is unloaded it will have more than enough strength to prevent dislocation during a seismic event.

The lifting pad eyes and the addition of the approved loading will not adversely affect the integrity of any poured concrete wall in the plant. This is based on the controlling installation document and discussion with NSEAS.

The lifting pad eyes will be used for maintenance of components that are in systems which are out of service, so the impact upon the plant is negligible. (SER 90-032-01)

37. MR 89-077 (Unit 1) and 89-078 (Unit 2), Crossover Steam Dump System. MRs 89-077/078 install temperature telltales to determine if the crossover steam dump 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 will not be impacted during a seismic event. The existing crossover steam dump system and the proposed gauges are not seismically qualified and are non-QA Scope.

There are no Appendix R concerns. This installation will not affect the operation of the steam dump system. (SER 90-033)

38. MR 89-091 (Unit 1), Refueling Cavity. The modification installs a permanent set of brackets on the wall of the lower refueling cavity of each unit. The brackets will hold an all-fiberglass ladder, which will be used for various reasons in the lower cavity. The ladder will be attached to the brackets when not in use. This includes at-power operation as well as during periods of refueling shutdown when the cavity is flooded.

Summary of Safety Evaluation: The ladder is modified to remove all aluminum and other materials which are not desirable in the borated cavity water or in a post-LOCA containment environment. The only remaining materials are: Stainless steel (small amounts on the ladder and the entire bracket); fiberglass (the ladder, including rungs); and natural rubber (the new ladder feet). The fiberglass and the natural rubber were analyzed for leachable chlorides and fluorides. Based upon these analyses, the materials will not have an adverse effect on the cavity water or on the reactor coolant system.

The total weight of the ladder is ~30 pounds. It is being rigidly held to the cavity wall using 3/8" flat stock. In the event of a design basis accident, the ladder will not break loose and become a missile hazard. Even if it would break loose, its physical location is in the bottom of the lower cavity, which negates any missile concern. Since fiberglass is heavier than water, it will not float in the event that the cavity is flooded up from containment spray or RCS break flow. Therefore, this ladder cannot become a hazard as a missile, nor will it impact the containment sump. The ladder will remain in the lower cavity.

The only possible accident which the installation could affect is a fuel handling accident. Due to the small size of the installation, and due to its significant distance from any fuel motion path, it is concluded that the modification will not increase the probability of a fuel handling accident. (SER 89-145)

39. MR 89-100 (Common), Reactor Makeup Water System. The modification abandons in place the "A" reactor makeup water tank, but keeps "A" reactor makeup water pump in service. This is being pursued to ensure positive control to prevent inadvertent use of the water in the "A" tank.

Summary of Safety Evaluation: The "A" tank is presently isolated from the rest of the reactor makeup water system because the water in it is out of specification (too much oxygen), and it would not be cost-effective to fix the tank bladder. The tank is not needed for continued operation because there is enough capacity for the

system in the "B" tank alone, and it has backup capability via direct connection to the DI water header.

The low-level trip function on the "A" pump will be changed to operate off the "B" tank level. The change utilizes the existing relay on the level channel for "B" tank, which also provides the trip function of "B" pump. This means that the failure of this one instrument loop (161) would disable both pumps. This failure, however, would be in the safe direction because it would only prevent dilution of the RCS, and does not impact any boration paths to the RCS.

Permanent piping changes will be performed per B31.1-1967. All materials will be stainless steel. (SER 89-127)

40. MR 89-101 (Common), HVAC. The change adds ventilation to the Unit 1 El. 66' fan room valve lapping cubicle room which will exhaust the fumes in the room so that the concentration will not exceed the threshold limits as specified by OSHA.

Summary of Safety Evaluation: The change consists of adding a fume exhauster for ventilation and a louver for air makeup. The system is designed to keep the concentration of the fumes generated below the OSHA threshold limit value. The system will operate whenever the room is occupied. Capacity of the system was calculated by calculating the needed ventilation for the room. The fume exhauster exhausts directly to the fan room, and makeup air will enter the lapping room through the louver. Calculations were performed to check the concentrations in the fan room with the exhausted air from the lapping room and the calculations show that the threshold limits will not be reached. (SER 91-044)

41. MR 89-127*A and IWP 89-127*A, Main Control Boards. MR 89-172*A isolates and replaces the existing instrument bus 2Y02 duplex receptacle (EXBA), which supplies the spray additive eductor suction valve (2SI-836B) control circuit, with four-outlet receptacle. An additional four-outlet receptacle supplied by the same feeder circuit will be installed in MCB C01 near receptacle MZB.

Summary of Safety Evaluation: Only one duplex receptacle supplied by instrument bus 2Y02 presently exists in main control board C01. Both outlets in this receptacle (labeled EXBA) are used to supply the spray additive eductor suction valve control circuit. One additional outlet supplied by instrument bus 2Y02 is required to supply the P-38B discharge valve hand controller (PC-4019) when MR 89-127*B is installed. MR 89-127*A will isolate and replace duplex receptacle EXBA with a four outlet receptacle. An additional four outlet receptacle, supplied by the same circuit, will be installed in main control board C01 near PC-4019. (SER 90-095-01)

42. MR 89-128*A and *B (Common), Emergency Diesel Generators. The modification installs isolation valves and test tees on several instrument lines to the G01 and G02 control panels (C64, C65). The installation will be done in a conservative manner.

Summary of Safety Evaluation: A calculation was performed to determine the maximum allowable span of 1/4" tubing with a 2-1/4" Whitey valve located anywhere in the span. The resultant maximum 5' span has been included in the work plan. Because of the seismic acceptability, the probability of failure is not significant. In addition, a worst case failure of the valves or tees would have no different impacts on diesel operability than a postulated failure of the existing tubing. (SER 89-101)

43. MR 89-129 (Unit 1), Instrument Buses. The hand controller for the Unit 1 accumulator vent valve and the hand controllers for the boric acid tank recirculation valves are presently plugged into a receptacle which is supplied by safeguards instrument bus 1Y03. The wires which feed this receptacle pass through another receptacle upstream in the circuit. This other receptacle is supplied by non-safeguards instrument bus 2Y06. MR 89-129 modifies the existing arrangement such that one receptacle is supplied by non-safeguards instrument bus 1Y06 while the other receptacle will be supplied by non-safeguards instrument bus 2Y06. Once installation is complete, the controllers will be verified functional.

Summary of Safety Evaluation: Instrument buses 1(2) Y06, although not considered safeguards instrument buses, are capable of being fed by the emergency diesel generators. They are not loads which would be automatically be stripped. A load calculation was performed to determine how much additional load could possibly be added to the diesels due to installation of this MR. The maximum additional load for G01 was determined to be 0.005 amperes. The maximum additional load for G02 was determined to be 0.00167 amperes. This is a maximum value because the added controllers are not in continuous use and, when in use, are not supplying maximum output. Therefore, the additional load can be considered negligible.

Installation of this modification will affect the receptacle which supplies the electric auxiliary feedwater discharge valve hand controllers (PC-4012 and PC-4019). These controllers will be supplied temporarily from a receptacle located in main control board 1C03. The receptacle in MCB 1C03 which will be used is also supplied from instrument bus 1Y03. (SER 91-016)

44. MR 89-143 (Unit 2), Buildings & Structures. MR 89-143 adds shielding around the regenerative heat exchanger cubicle to eliminate the high radiation area that currently extends into the hallway outside the cubicle. Temporary shielding in the form of vinyl covered lead blankets will be placed around the letdown line and around the gap above the door.

Summary of Safety Evaluation: No lead will come in contact with the stainless steel piping. Temporary shielding will be in place only during outages and will be administratively controlled.

Permanent shielding will be added to the cubicle door. The original design does not appear to have been installed in accordance with design control requirements. Thus, no documentation regarding material qualification or structural analysis is available. These modifications simply add four more 1/8" lead sheets (for a total of 3/4" thickness) to the existing frame. (SER 90-045)

45. MR 89-153 (Unit 1) and 89-154 (Unit 2), Secondary Sampling System. MRs 89-153/154 change out the existing secondary sampling system oxygen analyzers, and replace them with more reliable and accurate analyzers.

The new analyzer is a three-channel unit, which will replace the existing three single-channel units. The existing units output concentration levels and have a manual range selector switch. The output of the new analyzer will provide oxygen concentrations, temperatures and an auto-ranging function.

Summary of Safety Evaluation: The change will be a direct replacement. The oxygen cells will also be changed out, but the fittings on the new cells are directly compatible with the existing tubing in the sample panel. The modifications will not change the functionality of this system. (SER 90-026)

46. MR 89-182 (Common), Buildings & Structures. MR 89-182 installs a lockable high radiation area barricade gate in the blowdown evaporator cubicle. This barricade is necessary because: a) The expansion joint on the bottoms recirc pump has radiation levels >1000 mR/hr at 18". This triggers the use of a lockable high radiation area barricade or a barricade and a flashing warning light (per TS 15.6.11); and b) NRC IN 88-79 interprets our Technical Specification use of a flashing light to be strictly a short-term solution to the radiological protection requirements.

Summary of Safety Evaluation: The design of this barricade is in accordance with the requirements of 10 CFR 20, NRC IN 88-79, TS 15.6.11, OSHA, the Wisconsin Administrative Code, and the WE Safety Manual.

The barricade design is identical to the 16 lockable high radiation barricades installed under MR 86-048 in that they: a) Will close without personnel attention; b) will not block access to or egress from the bottoms recirc pump area of the blowdown evaporator cubicle; c) will not be constructed of combustible materials; and d) will be lockable. (SER 90-002)

47. MR 89-191*A (Unit 1), Residual Heat Removal System. The modification addresses the reconfiguration of the 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 discharged into Sump A. This design package decouples the relief valve discharge from the containment floor drain system, removes a section of the old relief valve discharge line to ensure adequate seismic clearance, and installs a temporary plug on the floor drain piping in the line.

Summary of Safety Evaluation: The change is being performed due to seismic qualification requirements for the relief valve and inlet piping configuration. The change allows qualifying the line for decay heat removal use until the piping configuration can be completely corrected.

The modification is being done to decouple the analysis forces of the floor drain system (non-seismic) from the seismic residual heat removal suction piping as transmitted through the relief valve.

The installation will be done to the requirements of B31.1, to the origina' Westinghouse design requirements and the requirements of modificatic equest M-46.. The entire revised configuration, including the 10" AC-601R-2 esidual heat removal piping, was reviewed for design loadings including pressure, deadweight, seismic and relief valve discharge. (SER 89-138-01)

48. MR 89-199 (Unit 1) and 89-200 (Unit 2), Building and Structures. The modifications install three platforms on the missile shield. The platforms are used to access the CRDM shroud fan suction ducting flanged connections.

Summ ry of Safety Evaluation: The CRDM shroud fan suction ducting is equipped with three flanged connections. Each of these connections will be provided with a removable access platform. Each platform is supplied with toeboards, handrails with openings for the ducting, post-DBA qualified paint, and is designed to Seismic Class 2 over 1 criteria.

The platforms must be removable to provide a free path for moving the reactor vessel head into/out of its lay-down area on El. 21'. The attachment to the missile shield will be made by a bracket, which extends somewhat beyond the edge of the missile shield. Based upon field inspections, this bracket will not present an

unacceptable obstruction to motion of the reactor vessel head into/out of the El. 21' lay-down area.

The platform's removal/installation at the beginning/end of refueling outages may be safely done in a manner similar to the removal of the CRDM shroud fan's ducting support brackets. This is controlled under procedure RMP 96. (SER 91-013-00)

49. MR 90-032 (Unit 1), Waste Gas System. MR 90-032 installs two check valves in the hydrogen line to the Unit 1 volume control tank (VCT), two check valves in the nitrogen line to the Unit 1 VCT, and a guide-type pipe support on the Unit 1 VCT hydrogen supply pipe. The purpose of this modification is to provide double-valve isolation between radioactive gas systems (cryogenics and VCT cover gas), and the hydrogen and nitrogen gas systems.

Summary of Safety Evaluation: Per FSAR Appendix A, the VCT and its associated piping is Seismic Class I. The nitrogen, hydrogen and cryogenics gas systems are Seismic Class III. Appendix A also states that, "The interface between a Class I and a lower class system is a normally closed valve or a valve which is capable of remote operation from the control room." For the VCT gas makeup system, a normally closed valve would be impractical and a remotely-operated valve is not supplied.

Regulatory Guide 1.29, "Seismic Design Classification," for Seismic Category 1 describes the interface as, "The system boundary includes those portions of the system required to accomplish the specified safety function and connecting piping up to and including the first valve (including safety or relief valve) that is normally closed or is capable of automatic closure when the safety action is required." For the VCT gas makeup system, the safety function that would be required is the isolation of the piping leading to the VCT to prevent the release of fission gases from the VCT to the auxiliary building.

Based upon RG 1.29 and the related discussion held during MSSM 90-04, the VCT gas makeup line check valve (1CV-263) constitutes the separation between the Seismic Class I and Class III piping. All portions of this modification will be installed in the Seismic Class III portions of the Unit 1 VCT gas makeup system piping. In addition, Seismic Class 2 over 1 criteria has been applied in the design. All material specified is compatible with the existing arrangements and is in accordance with Westinghouse Pipe Class 151R.

Failure of the VCT gas makeup piping is bounded by the safety analysis for a VCT rupture (FSAR Section 14.2.3). This modification will not increase the likelihood of a rupture of the VCT gas makeup system piping. (SER 90-061)

50. MR 90-038 (Unit 1) and 90-039 (Unit 2), Radiation Monitoring System. The modifications replace the insulation and heat tracing for the sample suction and return lines for RE-211/212. The heat tracing presently stops about 6" below the personnel hatch penetration. The present insulation is asbestos. Heat tracing and insulation will also be added around the penetration into the hatch.

Summary of Safety Evaluation: Once installed, the heat tracing will be tested to ensure correct operation. The circuit will be energized by adjusting the thermostat. The suction line thermocouple will be tested by verifying a temperature increase on the freeze protection recorder.

The load on the circuit breaker will be slightly increased due to the new heat trace cable. Based upon the size of the circuit breaker, the additional load can be considered negligible.

The modifications are QA-Scope because of seismic and containment pressure boundary considerations. By engineering judgment, the seismic adequacy of the sample suction and return lines will be maintained in view of the insignificant change in insulation weight. Following installation of the modifications, the load on the suction and return lines will be decreased due to the difference in weight between the fiberglass and present asbestos insulations. (SER 90-085)

51. MR 90-056*B&C (Common), Service Water System and IWP 90-056B. The design packages install temperature instrumentation for new battery room coolers HX-105A&B, and spent fuel pool heat exchangers, HX-13A&B, for monitoring heat transfer performance of the heat exchangers. This heat exchanger performance monitoring responds to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The new battery room coolers are being replaced under MR 87-159. Plant conditions required during installation are addressed in that modification.

Summary of Safety Evaluation: Thermocouples will be installed at the inlet and the outlet of the air side of the battery room coolers with local indication. The thermocouples will be attached to the support structure for the coolers. If the thermocouples break, the mass of the thermocouples will not impact the operation of the fans. The mass of the thermocouples is small in comparison to the mass of the fans.

RTDs will be installed in the service water lines for the battery room coolers and the spent fuel pool coolers. Both sets will have local indication. The RTDs will be placed in stainless steel thermowells. This construction allows removal of the RTD if necessary. The stainless steel is also resistant to corrosion which reduces the possibility of failure of the RTD.

The margin of safety is not diminished because the operation of the SFP heat exchangers and battery room coolers are not impacted by the installation of the instrumentation. (SER 90-114-01)

52. MR 90-056*D, Containment Ventilation System. The design package installs temperature instrumentation for 1HX-15D, the lowest containment fan cooler (CFC), for monitoring heat transfer performance of the heat exchangers. Performance monitoring of containment fan coolers is being performed in response to Generic Letter 89-13.

Summary of Safety Evaluation: Thermocouples will be installed at the inlet and the outlet of the containment fan cooler coils with local indication to be used for monitoring. The thermocouples will be attached to the structure containing the cooling coils. If the thermocouples break, the mass of the thermocouples will not impact the operation of the fans. RTDs will be installed in the service water lines for monitoring, these will also have local indication. If the RTD thermowells broke off in the system, they would be trapped at the bottom of the first vertical riser in the system and would not impact the flow in the pipe. The diameter of the thermocouples is 1/2" and they are 4.5" long. The pipes are 8" diameter so the flow will not be impacted by the installation of the RTDs. (SER 90-114-02)

53. MR 90-060 (Common), Buildings, Structures. MR 90-060 installs four Par-Kuts for security use (one at each containment hatch), provides them with 120 V AC power, and installs intrusion barriers at each containment hatch.

Summary of Safety Evaluation: Installation of the Par-Kuts supports conformance with 10 CFR 75.55(d)(8), which requires that a guard or watchman provide positive access control to containment during refueling or major maintenance activities. The 120 V AC power will be used for lights and heat in each Par-Kut. The barriers will block a potential intruder access point that exists when the equipment hatch shield blocks are removed. (SER 90-034)

54. MR 90-062*A (Common), Industrial Safety. A series of modifications address energy conservation commitments made to the Public Service Commission of Wisconsin (PSCW). This MR converts 122 exit signs at PBNP from the original incandescent lamps to fluorescent lamps in response to PSCW commitments.

Summary of Safety Evaluation: A number of the exit signs that are included within the scope of MR 90-062*A are powered from the emergency diesel generators (EDGs) through emergency lighting panels. Therefore, this modification results in decreased electrical demand on the EDGs. (SER 91-007)

55. MR 90-074-01, Condensate System. MR 90-074-01 adds a pressure tap on a condenser steam dump header, downstream of the dump valve in order to get accurate pressure drop readings across the valve. Pressure drop data will be used to verify dump header flow rates for doing MR 90-074/075*B, which add holes in the header to decrease orifice exit velocities.

Summary of Safety Evaluation: Installation of the addendum requires that one of the eight condenser steam dump headers be isolated. This reduces the capacity of the condenser steam dump system to 35% load. If this system actuated during installation, the only effect on the RCS would be slightly higher Tavg, or slightly slower cooldown rates. Other redundant systems would still be available to augment this function (rods, power operated reliefs, etc.).

The installation could also affect condenser vacuum. This effect will be minimized by the size of the hole (1/8") and actual installation method/planning (prefabricated valved connection, weld to pipe, drill down through centerline of connection, shut valve as drill is being withdrawn).

Some metal shavings will enter into the condenser during installation. Due to the small size of the hole (1/8"), only very small, thin shavings will be produced. This will not adversely affect the condenser, condensate or feed pumps, or steam generators.

The connection up to and including the root valve will be designed, installed, and tested per B31.1. Failure of the gauge would not impact the safe operation of the plant since it would be normally isolated from the system. Consequences of gauge failure with the root valve open, are limited by the size of the hole (1/8"). Steam flow is choked across the dump valve, and therefore, an 1/8" opening downstream will not increase steam flow in the header. (SER 90-088-01)

MR 90-074*A (Unit 1), Circulating Water System. The design package installs impingement grating which deflects turbine steam exhaust, thereby protecting the tubes from steam erosion. Installation will be at locations where tubes are identified as being eroded by turbine exhaust. The general location of the grating will be near the top of the tube bundles and along the outer tube bundle perimeter.

Summary of Safety Evaluation: Installation of the modification is an effort to control steam erosion of the condenser tubes. The purpose is to prevent a tube rupture which would require a costly repair, and also to extend the tube life. The design involves the installation of impingement grating which will deflect turbine steam exhaust and thereby protect the tubes from erosion. Installation will require that the plant be in a cold shutdown condition, so it will occur during a normally scheduled refueling outage. The grating will be located where tube erosion has been identified by both eddy current and visual inspections.

The turbine exhaust will deflect off the impingement grating instead of the tubes as was the case in the past. All materials are of 304 stainless steel for strength and effective erosion resistance. 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. (SER 90-088-02)

MR 90-074*B*C (Unit 1) and MR 90-075*B*C (Unit 2), Circulating Water System. Design Package B involves drilling additional holes in the longest of the four bypass steam dump lines in each condenser. The purpose is to increase the steam dump distribution header flow area and thus reduce steam velocity in the pipe. The reduction in steam velocity will reduce erosion of tubes near the bottom of the bundle. The volume of steam passing through the pipe is controlled by a system pressure signal from the steam bypass valves. The flow volume is also limited by the pipe size as documented in the FSAR, so the additional holes do not significantly alter the operation of the bypass steam dump. The design package also does not affect the seismic qualification of the bypass steam dumps or the condenser (Seismic Class III per Appendix A of the FSAR).

Design Package C reverses the modulating sequence of the condenser steam dump valves so the longest operating time will be assigned to the physically longest condenser steam dump diffusers. This effort will remove the erosion source from the immediately affected area. Reversing the sequence will be achieved by exchanging respective I/I repeaters inside rack 1(2)C107. The I/I units will be relabeled and then calibrated per applicable procedures.

Design Package C also rearranges the condenser valve position indication lamps on the front of 1(2)C03. The lamps presently sequence open from left to right on Unit 1, and from right to left on Unit 2. This sequence will be retained. The alphanumeric sequence (right to left for Unit 1) will, however, be sacrificed in favor of maintaining the display consistent with operator expectations. Lamp rearrangement will meet the mirror imaging and display integration requirements of the control room design document, DG-G01.

Summary of Safety Evaluation: Concern for increased probability for a condenser tube rupture due to this modification has been considered. The additional holes to be drilled in the bypass steam dump pipe are of the same diameter, spacing, and pattern as the original Westinghouse design. Therefore, stress remains evenly distributed per unit length of pipe.

The "blow-open" features of the condenser steam dump system will not be affected by this modification. Seismic qualifications of the main control boards will not be affected, and there are no Appendix R concerns since all rewiring efforts will occur within the control boards. Note that new SIS type wire may be used where the existing SIS wire is too short to accommodate lamp rewiring.

The safety concerns from an FSAR accident analysis viewpoint are not applicable. This is because the condenser is not safety-related, and its heat removal characteristics are not taken credit for in the FSAR. (SER 90-088)

56. MR 90-076 (Unit 1) and 90-077 (Unit 2), Auxiliary Feedwater System. MRs 90-076/077 provide a simple method of checking to see if the drain line in the 1(2)P29 auxiliary feedwater pump exhaust line is clear. The test is desired to prove that the line is clear before cold, fast start testing of the Terry turbine.

Summary of Safety Evaluation: The modification adds a tee/valve assembly to the drain line of the exhaust line. The tee will be used for connecting a water source to the drain line for flushing purposes. The valve will be used for isolating the drain line during testing. A seismic support is added to the drain so it will meet Seismic Class 1 ratings for small bore piping. This design provides a simple method of testing and meets the applicable codes. The function of the auxiliary feedwater system will not be altered. (SER 90-091)

57. MR 90-086*A (Unit 2), Buildings and Structures. MR 90-086*A adds handrails and several small platforms to the Unit 2 steam generator shield walls at El. 76'. Handrails do not presently completely enclose these shield walls, and the existing platforms do not conform to OSHA standards.

Summary of Safety Evaluation: The platform and handrails were designed to meet Seismic Class 2/1 criteria as they are located over safety-related components in the reactor coolant and residual heat removal systems. Equipment seismic and thermal clearances have been included in the design. State certified structural welders will perform the fabrication. Paints specified are compatible with the post-DBA containment atmosphere. The design also conforms with OSHA specifications, as required by the FSAR and federal law.

Steam generator cubicle LOCA blowdown forces are not included as a design input (for prevention of missile generation) as the leak-before-break philosophy has been accepted by the NRC. (SER 91-078)

58. MR 90-103 (Common), Buildings & Structures. MR 90-103 replaces some of the ceiling tiles with aluminum eggcrate lay-in panels in the control, computer, and instrument rack rooms.

Summary of Safety Evaluation: FSAR Section 7.7.5, "Control Station Layout, Information Display, and Recording," described the control room ceiling. The replacement of these tiles will not affect fire barriers. (SER 90-069)

59. MR 90-106 (Unit 2), Chemical and Volume Control System. The MR provides a common hard pipe vent path from the hydrogen regulators, 2PCV-113 and 2PCV-158, and the nitrogen regulator, 2PCV-114, to the VCT cubicle exhaust duct.

Summary of Safety Evaluation: Current accepted practice for adjusting hydrogen regulators, 2PCV-113 and 2PCV-158, and nitrogen regulator, 2PCV-114, vents gas to the PAB. The vented gases are picked up by the auxiliary building exhaust header and flow past the vent stack monitors before being discharged to the outside atmosphere. The installation of a hard pipe vent line from the current vent points to the VCT cubicle exhaust duct will only give the vented gases a direct path to the exhaust header, via a pair of isolation valves. There will be no change in the gases being vented (amount, type or radiological activity). All new materials are the same type as those used at the CVCS tie-ins; 3/8" stainless steel tubing.

Swagelok fittings, and valves; and will meet the specifications of the components to which the tie-ins are made, Westinghouse piping Class 151N. (SER 91-076)

60. MR 90-107 (Unit 1) and 90-108 (Unit 2), Electrical Distribution System. MR 90-107/108 install 3 test points and 2 test switches in the undervoltage circuit for each of the 1(2)A01 and 1(2)A02 buses. This is being done so the time delay relays used for stripping the buses can be timed.

Summary of Safety Evaluation: For each of the buses, one test switch will prevent the bus stripping relays from energizing. This will allow testing of the undervoltage relays while the unit is at power.

The other test switch will be installed in the trip circuit for the station service transformer breakers. This will allow us to test the stripping of the bus during a refueling outage, since the station service transformer breakers are the only breakers on the A01 and A02 buses that will be closed.

The test points will be used to verify contact operation and coil integrity during testing. They will also be used to verify that the knife switches properly close upon completion of the testing. The test points will be added to the existing undervoltage test panel on each of the buses. The cutouts required for the test points are small and the mass of the test points is negligible and therefore will have no structural impacts on the bus.

Installation will be done during the respective unit's refueling outage. During installation, the undervoltage circuit for each bus will have to be deenergized. This will generate a reactor trip signal if the buses are done simultaneously.

Since work also has to be done to the station service transformer breaker trip circuit, it too will have to be opened. To do this the 1(2)B01 buses will have to be temporarily resupplied from the tie breaker to the 1(2)B03 buses and the 1(2)B04 buses will have to be temporarily resupplied from the tie to the 1(2)B04 buses. A temporary jumper will have to be installed to allow closing the tie breakers in on a live bus. Double verification will be used to verify that the jumper is properly installed and removed. Section 8.2-4 of the FSAR describes these tie breakers stating that they utilize a dead bus transfer scheme, because no synchronization ability has been provided. Since the two 480 V buses will be supplied from the same source, no synchronization is required and the tie breakers can be closed in on a live bus. (SER 90-089)

Summary of Safety Evaluation: The control fuses for the breakers are being removed in order to prevent the breakers from closing. The installation procedure for this modification requires that these breakers be closed to temporarily supply the B01 and B02 buses through the tie breakers. In order to do this, the control fuses for the tie breaker will be reinstalled to close the breaker. Once the breaker is closed, a cable lead in the B03/B04 cubicle will have to be lifted to prevent the breaker from closing again if it should trip open. Since this breaker will trip open on B03/B04 undervoltage, the breakers will open prior to the diesel starting.

Closing of these tie breakers is not a safety-related function. Tripping these breakers, which is safety related for stripping B03/B04 on undervoltage or SI, is not being effected by lifting the wire.

Once installation is complete, the control circuits will be returned to the as-found condition, with the wire reterminated and the control fuses pulled.

The installation will be coordinated such that no more than one tie breaker will have its control fuses installed without having the lead lifted. This will prevent a single failure from effecting both trains. (SER 90-089-01)

61. MR 90-113 (Common), Fuel Oil System. The modification provides seismic support of the fuel oil transfer piping in the pumphouse. A support will be added to the disconnected pumpout piping to limit its displacement during a seismic event. A spoolpiece will be utilized to reconnect the pumpout piping to the main system if pumpout of the tank is required.

Summary of Safety Evaluation: An existing support will be replaced by one which provides horizontal restraint and ensures the piping will remain operable following a seismic event. Additionally, the pumpout piping will be disconnected on the seismic portion.

The modification provides an installation consistent with FSAR commitments and ensures availability of the fuel from the emergency tanks to the diesels. (SER 90-042)

62. MR 90-118 (Unit 1) and 90-119 (Unit 2), Containment Spray System. The MRs install a test connection in each containment spray pump suction line. The test connection consists of a 1/2" tap off of the suction line, a 1/2" isolation valve and a 3/8" tubing connection downstream of the valve. A test connection will be installed between valves 1(2)SI-858A and 1(2)SI-870A and between valves 1(2)SI-858B and 1(2)SI-870B. The test connections are being added in order to facilitate the leak testing of valves 1(2)SI-870A and 1(2)SI-870B.

Summary of Safety Evaluation: The test connections consist of a 1/2" tap, 1/2" valve and a 3/8" tubing connection with end cap installed off the top of the 6" suction lines. The test connection consists of welded construction up through the isolation valve with threaded and Swaged connections downstream of the isolation valve to facilitate testing. The test connection valve will be shut during normal operation and will be included on the system checklists.

The test connections were designed and will be installed in accordance with original construction and design specifications. The configuration has been evaluated and will not affect the seismic qualification of the suction piping.

The test connections will be installed when the affected unit is in the cold or refueling shutdown condition. In this plant condition, the containment spray system is not required to be operable per the Technical Specifications. The installation work plan has specific cleanliness controls so foreign debris or materials do not enter the suction piping. (SER 91-010)

63. MR 90-122 (Unit 1) and 90-123 (Unit 2), Chemical & Volume Control System. The modifications replace the CV-111 lower limit switch. The present limit switch will not actuate until the valve has stroked enough to allow as much as 60 gpm of flow through it. Actuation of the switch is important in that it brings in the "potential dilution in progress" alarm besides providing valve position indication.

Summary of Safety Evaluation: The new limit switch will be mounted to a steel plate which, in turn, will be mounted to the existing limit switch bracket. This mounting is necessary because of the size difference between the old and new switches. Based upon the weight difference between the old and new switches, the seismic adequacy of CV-111 will be maintained. The existing limit switch

wiring is adequate and will be used for the new lower limit switch. Once installed, the valve will be cycled to ensure correct limit switch operation.

The modifications are non-QA Scope. Although CV-111 is QA due to seismic and pressure boundary considerations; the modifications are non-QA because the limit switch is not a functional part of the boundary. (SER 90-084)

64. MR 90-129 (Unit 1) and 90-130 (Unit 2), Service Air System. MR 90-129/130 change out the SA-17 valve on each unit. This is the service air containment isolation valve outside containment. It is a swing check valve and has had a history of trouble passing leak tests. The modifications change it to a gate valve. It will also be necessary to install a tee upstream of SA-17 which will contain a flange. This will be to provide a connection point for future ILRT bleed downs. IVTCs will be modified as necessary for ORT testing.

    Summary of Safety Evaluation: The modification is expected to decrease probability of a failure of containment integrity. The existing valve is a check valve with a known history of leakage. The new gate valve is a Powell gate valve with a flexible wedge design. It is anticipated that the leakage will be decreased significantly.

    The modifications will be installed with the unit in the cold or refueling conditions when containment integrity is not required. The first off valves inside containment will be red tagged closed during installation to assure containment closure is maintained if we need to move fuel or conduct other refueling operations during the course of installation. (SER 91-011)

65. MR 90-141 (Unit 1) and 90-142 (Unit 2). Buildings and Structures. The MRs provide a safe storage location for the steam generator crevice flush vacuum hoses. The hoses were laid on the walkways (El. 85') between the "A" and "B" MSIVs. This creates a walking hazard and reduces the free space for transportation of relief valves during an outage.

    Summary of Safety Evaluation: The design consists of welding brackets to the walkway handrailings. The design was analyzed to verify the load capabilities of both the brackets and the handrailings with the brackets attached and hoses in place. The analysis indicated that the brackets can support the hoses, and also the handrailings can support the brackets and hoses. The design also took into account that the walkways are rated as Seismic Class 1 by determining that the additional loading incurred by the hoses will not affect this rating. A safe storage area is provided for the hoses because the hoses are no longer stored on the walkway grating but supported above the grating which yields a clear walkway and transportation area. The applicable codes for this design were addressed and are fulfilled by the proposed design. (SER 91-037)

66. MR 90-152 (Unit 1), Main Steam System. MR 90-152 eliminates the potential of having a failure of a contact block affect both trains in the main steam isolation valve (MSIV) manual initiation circuit. The modifications were initiated as a result of an investigation in response to a vendor letter dealing with potential common mode failures due to the use of single OT2 type switches in safety-related systems.

    Summary of Safety Evaluation: Failure of a contact plunger to be depressed is a single failure that would affect the entire switch because it would prevent the switch actuator from turning. This failure would affect the opening of the MSIVs, which is not a safety-related function. Also, this failure would be immediately detected since the operator would not be able to turn the switch.

The other type of failure is the failure of a contact plunger to reset once it is released. A contact failing to reset will also affect those contacts mounted directly behind it, but will not affect those contacts on the other side of the switch. Because of this, the contacts are being rearranged so A train contacts are on one side and the B train contacts are on the other side. This is given an alternate solution if control board space is a problem.

The modifications will have no system functional effect on the MSIVs as described in FSAR Section 10.2.2 or their control circuit as described in Section 7.2.2. The requirements of manual actuation of safety-related systems, per Section 7.2.1, are being maintained. The change has no impact on the automatic main steam isolation circuitry. (SER 90-080)

67. MR 90-154 (Unit 1), Main Steam System. MR 90-154 replaces the MSR stilling manifold drain lines with SS pipe and fittings. Ultrasonic testing has shown a trend of wall thinning with a few locations getting very near the minimum Code wall thickness.

Summary of Safety Evaluation: The material selected for the replacement pipe has better resistance to corrosion/erosion, and also has a slightly higher allowable stress. The change is a straight replacement, using the same size and schedule of pipe, and the same supports and support locations.

The piping system will be designed, installed and tested per B31.1-1967. This will provide adequate assurance that the piping system will not fail under operating conditions. (SER 91-017)

68. MR 90-161 (Unit 1), Fuel Transfer System. The modification installs lead bricks overlapping the expansion joint between the fuel transfer canal structure and containmen[illegible] side the canal structure. Thirty lead bricks, 2"x4"x8", will be stacked v[illegible]y and seismically supported.

Summary [illegible] Safety Evaluation: The gamma ray scatter from the fuel transfer tube will be reduced with lead sheets stacked on the fuel transfer penetration tube. Twenty 1/8" layers (accumulated thickness of 2.5") will be strapped individually with 0.75" wide, 0.030" thick stainless steel bands on each end. The last layer will be secured with straps positioned every 3".

With the additional lead shielding, no accident previously evaluated in the FSAR will be affected. Furthermore, the lead will not impact the probability and consequences of a malfunction to equipment important to safety previously evaluated in the FSAR. The probability of an accident or malfunction of equipment important to safety different than those mentioned in the FSAR is highly unlikely due to improbable required physical or natural accelerations. Since the lead has no bearing on the function of the fuel transfer tube, the margin of safety per the Technical Specifications will not be affected. (SER 90-077)

69. MR 90-166 (Unit 1), Fuel Transfer System. MR 90-166 installs a handrail to the Unit 1 manipulator crane motor platform (MCMP) along with a gateway to access the platform from the manipulator crane controls platform. Currently the MCMP is accessed by crawling through the fixed handrail of the MCMP and work is performed on the MCMP while wearing fall protection.

Summary of Safety Evaluation: General Design Criterion 2 requires that systems important to safety shall be designed to withstand the effects of earthquakes. Appendix A of the FSAR indicates the manipulator is Seismic Class III. The addition

of a handrail will not alter this seismic classification. Since the main danger is that the handrail could fall off the manipulator into the reactor vessel during fuel handling, the handrail was considered "important to safety". An analysis was done to verify that the handrail will not fall off of the manipulator into the refueling cavity during a seismic event.

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety.

Rupture of one complete outer row of fuel elements in a withdrawn assembly is assumed as a conservative limit for evaluating the environmental consequences of a fuel handling incident. Even in the unlikely event that the handrail should fall and strike one or more fuel assemblies, there is very little chance that more than the equivalent of one complete outer row of fuel elements could be ruptured. The postulated accident would be within the accident evaluated in the FSAR Section 14.2.1. (SER 90-076)

70. MR 90-168 (Unit 1), Reactor Coolant System. MR 90-168 removes the second-off isolation valve (RC-500C) from the reactor vessel level indication system (RVLIS) reference leg and replaces it with tubing. The valve, which is located in the refueling cavity, is not used for operation or maintenance of the RVLIS and is not required by B31.1-1967. Removal of the valve eliminates the need to check its position when performing CL-4A, "RCS Valve Checklist."

Summary of Safety Evaluation: The modification will be installed during a refueling outage when operability of the RVLIS is not required. The valve will be replaced with stainless steel tubing and fittings of the same type as those already existing in the RVLIS reference leg. The tubing and fittings have pressure and temperature ratings in excess of the design ratings of the RCS. The new components will be seismically supported. The new components will be checked for leaks in accordance with TS 15.4.3 during the post-refueling pressure test of the RCS. (SER 90-103)

71. MR 90-178 (Unit 1), Primary Plant Instrumentation. MR 90-178 removes unused ΔTavg I/I 1TM-405F. The I/I was originally installed to provide a ΔTavg signal to the reactivity computer and was subsequently abandoned in place.

Summary of Safety Evaluation: 1TM-405F receives its input from summing amplifier 1TM-405D. 1TM-405D also supplies a ΔTavg signal to the rod insertion limit computers, the ΔT deviation alarms on 1C04, and to the PPCS. Removal of 1TM-405F will not change the electrical characteristics of the instrument loop since the 100 ohm input loop will be temporarily de-energized while the 1TM-405F signal cord is removed from the terminal block. This interim condition is acceptable due to the short period of time required to remove the signal cord, and since the PPCS provides a separate rod insertion limit alarm. (SER 91-009)

72. MR 90-183 (Unit 2), Feedwater Line Seismic Support Upgrade. The modification modifies or removes a number of supports on the Unit 2 main feedwater supply lines (both "A" and "B" loops). These modifications are necessary to upgrade the support of the feedwater piping in the Seismic Class 3 portion of the feedwater system in order that the Seismic Class I portion of the system at the 1/3 interface (second-off check valves 2CS-476AA and 2CS-466AA, outside containment) are not affected by seismic movements.

The changes are necessary because the structural attachments of the supports to existing embedded plate steel was inadequate, not from an operability standpoint, but from compliance with B31.1-1967 and AISC Sixth Edition requirements and to maintain containment integrity for the feedwater lines. When complete, both pipe stress and support load code requirements are met.

Summary of Safety Evaluation: The modification upgrades the supports for a section of main feedwater piping just upstream of valve 2CS-466AA (for the "A" loop) and 2CS-476AA (for the "B" loop). (Refer to FSAR Figure 10.2-2A, Condensate and Feedwater System P&ID, and Bechtel Drawing P-211, System Isometric, Feedwater Loop A and B). These second-off check valves outside containment represent the boundary between the Seismic Class 1 and Seismic Class 3 portions of the feedwater system. The modifications to the supports upgrade this portion of the feedwater system to be in compliance with USAS B31.1-1967 (as specified in FSAR Table 10.1-2), and AISC, Sixth edition, codes and to protect the Seismic Class 1 portion of the system, downstream of the check valves to the containment penetration, from excessive influence from the Seismic Class 3 portion. FSAR Appendix A discusses the Seismic Class 1 criteria. FSAR Section 10.2.2 briefly discusses the feedwater system. The Class 1 portion of the system, including two check valves in series, serves as a containment isolation boundary for each steam generator.

The modification, installed under IWP 90-183, is performed with Unit 2 in the cold shutdown condition. The sequence stipulates modification of supports EB-9-H200, EB-9-H201, EB-9-2H11, EB-9-2H16, and EB-9-H202 and removal of supports EB-9-2H12 and EB-9-2H17 to be performed in any order, the order of modification being inconsequential. All materials are specified as QA Scope. Ignition control permits are required and are stipulated in the IWP. Field walkdowns and pre-installation discussions with the workers and between the installation group, the testing group, and the acceptance group are stipulated in the IWP. The requirements for temporary supports and/or rigging to support installation are also stipulated. Slight movements of the main feedwater piping will be necessary to allow modification of the supports; these slight deflections have been evaluated as safe, producing only slight and negligible stresses in the piping. Appropriate NDE (VT) is specified in the IWP to verify weld Code compliance. (SER 91-091)

73. MR 90-189 (Unit 1) and IWP 90-189, Control Boards and Racks. To eliminate violation of electrical separation requirements at the miscellaneous relay rack (1C158), three "B" train safeguards circuits are removed from the rack. This is accomplished by supplying auxiliary relays in main control board 1C03.

Summary of Safety Evaluation: Where the existing relay is not also providing a "A" train circuit function, the relay is relocated to the main control board. Where the existing relay is providing a Train A circuit function, a second relay is installed in the main control board and the coil is connected in parallel with the existing relay. Rewiring in the control board is done to assure train separation. The three affected circuits are as follows: a) 1P11B, CC pump; relay provides an automatic start of the pump on low CC header pressure; b) 1RC-427, RCS letdown isolation valve; relays close the valve on low pressurizer level; and c) 1RC-430, pressurizer PORV; relays open the valve on high pressurizer pressure.

When the modification is complete, the circuits will be functionally the same as before. Equipment location and arrangement has been changed to provide train separation. The coil portion of the circuit which is not train or safety-related is changed in some cases to have 2 relay coils in parallel. This is a typical design and should have no affect on circuit operability.

The installation will be controlled by IWP 90-189. Installation will occur during col shut down of Unit 1. The portions of the circuits to be modified will be isolated by opening sliders in the main control board. This allows the circuits to remain operable except for the specific control functions listed above. The 1P11B CC pump will be designated as the running pump and therefore, the loss of its autostart feature will have no affect on plant operability. Also, the low temperature overpressure control function for 1RC-430 will not be defeated except briefly during testing.

Work on the three identified circuits brings the plant into conformance with the instrumentation and protection circuit descriptions of FSAR Chapter 7. That is, to achieve physical separation of redundant Trains A and B. Although the specific circuits are not described in the FSAR, the separation criteria is. (SER 90-108-01)

74. MR 90-191*A (Unit 1), Safeguards Racks. NCRs N-90-073, N-90-074, N-90-075, and N-90-076 detail electrical separation conflicts in the safeguards racks. That is, A train circuits are in B train racks and B train circuits are in A train racks. These conflicts violate the train separation requirements of FSAR Section 7. The conflicts mainly involve containment isolation valves. In cases where there is a single active containment isolation valve, the one valve receives containment isolation signals from both trains of safeguards. Modification Request 90-191 eliminates these electrical separation conflicts in the safeguards racks.

Summary of Safety Evaluation: The design package involves using spare contacts on existing CI relays in the safeguards racks to supply auxiliary CI relays in the main control board. Wiring in the main control board is done in a manner to provide for electrical separation between circuits. These modified CI valve circuits are functionally the same as the original circuits. That is, either train of containment isolation actuation will cause the valve to close.

The addition of the new auxiliary relays adds one additional component to the actuation circuitry which could potentially fail. It is felt, however, that the probability of this is remote. Also, with these particular circuits, multiple failures would be required to block the containment isolation function. It should be noted that use of auxiliary relays is a typical design method which is used for other safety-related functions, such as stripping the battery chargers on SI.

There is no credible failure mechanism associated with the alarm relays which could disable the safety function of these circuits. The alarm relays significantly improve the reliability of these circuits by providing an alarm if the power supply is interrupted. (SER 91-029)

75. MR 90-195 (Common) & IWP 90-195, Electrical Distribution. The MR adds a new 13.8 kV box H01 voltmeter and relocates a gas turbine G05 output voltmeter in C02. These meters are supplied form potential transformers (PTs) at H01 and G05.

Summary of Safety Evaluation: None of these circuits have any safety-related functions. Local fuses at H01 provide isolation from the remote (C02) circuits. Therefore, failure of the new circuit or meter could not result in a loss of offsite power. The gas turbine voltmeter circuit will remain the same except for relocation of the voltmeter. This circuit is not currently fused in a manner that would isolate the local controls from the indication and controls. Thus, a failure of the gas turbine voltmeter circuit could disable the operation of the gas turbine. However, this failure would be readily detected and corrected.

IWP 90-195 controls installation of this modification. The circuits are isolated by removing fuses. This removes the gas turbine from service. Applicable TSs include 15.3.7.A.1.b, 15.3.7.A.2.b, and 15.3.7.B.1.b. Under normal conditions, loss of the gas turbine will have no effect on plant operations. These TSs would only be applicable if there was a loss of one or both of the high voltage station auxiliary transformers. If this were to occur during the outage of the gas turbine, one or both of the Point Beach units would have to be taken to hot shutdown. The plant condition required for performance of the IWP is that normal offsite power is available for both units. It is anticipated that the IWP will take two days to complete.

The NRC SER on the station blackout rule dated October 3, 1990, along with our response to the SER dated November 8, 1990, were reviewed and were found to not affect this installation. (SER 90-123)

76. MR 90-205 (Common), Buildings and Structures. MR 90-205 installs a bird barrier net over the circulating water systems crib intake structure to prevent the ingestion of birds into the system. The net will consist of a lightweight polypropylene mesh net. The net will be supported be steel cables spanning the intake opening. The net will be firmly secured to both the cables and the intake structure using bundle ties. The barrier net is designed for installation in spring and removal in the late fall for storage and is capable of being reused over a period of several years. The cable and net support system have been verified to be capable of supporting any additional weight due to ice formation and birds roosting on the net.

Summary of Safety Evaluation: The addition of the barrier net does not have any adverse affects upon equipment or systems that is important to safety. Even in the event of a catastrophic failure of the net, it would not prevent circulating water, service water and fire water from being supplied to the plant.

In the event that the barrier net did collapse and fall into the crib intake, the net would be prevented from entering into the circulating water, service water and fire pump suctions by one of the following means (all upstream of the pumps): a) The net would become entangled on the intake piping's butterfly valves; b) the net would be stopped by the installed trash racks which are specifically designed to prevent foreign material from entering into the circ. water pump suction; or c) the net would be stopped by the traveling screens. (SER 91-012)

77. MR 90-214 (Unit 1) and 90-215 (Unit 2), Drains. MRs 90-214/215 change out the containment sump "A" drain cover. The new cover has a removable lid with attached screen to allow for periodic cleaning of the sump screen.

Summary of Safety Evaluation: The new drain cover flow area and screen density are approximately the same as for the cover which will be removed from service. The new drain cover assembly is made entirely of stainless steel materials. The new drain cover is designed and will be installed to facilitate more complete drainage of the sump area. (SER 90-104)

78. MR 90-225 (Unit 1) and 90-226 (Unit 2), CVCS. MRs 90-225/226 replace volume control tank level transmitters 1(2)LT-112 and 1(2)LT-141. Transmitters 1(2)LT-112 and 1LT-141 are being replaced due to difficulties in calibration and repair efforts. Transmitter 2LT-141 was replaced during U2R16 via MR 90-226*A and uses a propylene-glycol fill fluid. 2LT-141 will be replaced so all four VCT level transmitters utilize a deionized water fill fluid (propylene-glycol has a slightly longer response time).

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

The modifications do not result in an increased probability of a rupture of either volume control tank, nor do they alter the VCT rupture analysis found in Section 14.2.3 of the FSAR.

There is no net additional load on the inverters or on the station batteries due to these modifications. Existing instrument bus separation is maintained. (SER 90-112-01)

79. MR 90-240 (Common), Fuel Oil Pumphouse Dike and Drain. The modification installs a step across the entrance to the lower FOPH to prevent fluid from draining to the lower FOPH. All floor penetrations through the upper FOPH to the lower FOPH will be grouted closed. Six 8" by 16" drains will be installed in the upper FOPH to provide adequate drainage capability. The drains are designed to retain heat in the FOPH in the event of failure of the FOPH heaters.

Summary of Safety Evaluation: The modification to the upper fuel oil pumphouse involves only structural modifications to the building, and does not impact the design or operation of the fuel oil transfer pumps or buried storage tank, T-72. The modification does not significantly weaken the upper FOPH structure or increase the probability of its failure. Installation of the dike and drains serve to protect the fuel oil transfer pumps from a flooding event in the upper FOPH and help to ensure the operability of the safety-related fuel oil transfer pumps.

The modification is scoped as requiring quality assurance surveillance. The material and fabrication are controlled in accordance with applicable standards. (SER 91-059)

80. MR 90-243 (Unit 1), Supports. The modification removes the existing oiler air line tubing support attachment, which was connected to the Unit 1 facade elevator structure, and constructs a support which is mounted to the Unit 1 containment structure.

Summary of Safety Evaluation: This modification prevents potential differential motion between the purge supply valve and the elevator structure from damaging the oiler air tubing. The purge supply valve is mounted to containment, and therefore, attaching the tubing support to containment prevents differential motion from occurring.

The modification mounts the support to containment using two 3/8" diameter Kwik bolts. The maximum expected penetration into the containment structure is approximately 2". This is not considered a significant degradation of the 3'6" containment wall thickness. No reinforcing will be cut.

The loading being carried by the containment structure will not affect its ability to perform its design function. Under normal conditions, the tubing is supported by the purge supply valve, which is attached to containment. Under accident conditions, the seismic loading carried by the containment through the support is low. Oiler weight is 1.7 lbm undergoing a maximum acceleration of 3.4 g per calculation N-91-029.

No change in the configuration of the purge supply valve or its air tubing will occur. The present tubing support will be mounted to containment rather than the elevator structure. (SER 91-023)

81. MR 90-244 (Unit 2), Heating and Ventilation System. This modification relocates containment recirculation fan roughing filter and cooling coil differential pressure gauges (DPI-4837 and DPI-4833) currently attached to the "B" containment ventilation system ductwork. These gauges are used to locally indicate the differential air pressure across the containment ventilation fan roughing filter and cooling coil during periodic inspections. Access to the gauges requires personnel to enter a high neutron field in order to reach them. To reduce personnel exposure, the gauges are relocated to the structural steel below the "B" ventilation ductwork. The gauges are then reinstalled similar to the original installation using copper tubing and brass fittings. Work is done during a normal refueling outage or at a cold shutdown when the ventilation fans are not required and containment may be safely accessed.

Summary of Safety Evaluation: The containment ventilation system is discussed in PBNP FSAR Sections 5.3, 6.3, 6.4 and 14.3.4 and in TS Sections 15.3.3 and 15.4.5. No reference is made, however, to the pressure indicators other than showing their existence in FSAR Figure 5.3-1, Sheet 2. Thus, since relocating the gauges does not affect the function served by these gauges, no changes to the FSAR will be required.

The modification maintains the original acceptable design for connecting the manometers. The manometers are securely attached to the structural steel below the ventilation ductwork in a location free from other fragile or vital equipment. Containment compatible materials have been specified for use to ensure that post-accident conditions are not aggravated (FSAR 5.6.2). After installation, the system will be equivalent to the original with no adverse affects introduced. (SER 91-067)

82. MR 90-247 (Unit 2), 120 V Electrical System. MR 90-247 removes the duplex outlet receptacles and their associated wiring which are located in the terminal block area of the Unit 2 NIS cabinets.

Summary of Safety Evaluation: The Unit 2 outlets are powered by the yellow instrument bus. They present the potential for unplanned load to be added in the event of their use. The modification reduces the probability of loss of the NIS Channel D because it removes outlets and wiring which could possible be a point of failure due to being overloaded or short circuited. (SER 91-080)

83. MR 90-269 (Unit 1) and 90-270 (Unit 2), Main Control Boards. MR 90-269/270 interchange the wires to the subcooling meters, selector switches and the indicating lights. The modifications are the result of CRDR HED 311. Presently, the subcooling meter for the "A" loop is on the "B" loop side of the ASIP and the "B" loop meter is on the "A" loop side. The subcooling meters are only loop associated when the loop RTDs are used for the subcooling margin input. However, the incore thermocouples are normally used for the subcooling margin input.

Summary of Safety Evaluation: The existing wires to the subcooling meters will be isolated, removed and discarded. New wires will be pulled as additional length is required because of the routing necessary to maintain train separation. Train separation for the subcooling meters is required because they are related to nuclear safety (Class 1E). The minimum separation required per the technical requirements for the ASIPs is 6" or an installed barrier. This minimum separation will be maintained

Following installation of the modification, the subcooling meters will be calibrated to ensure that they are functional and within tolerance. The selector switches and their associated lights will be verified functional by toggling between "RTD" and "TC."

The modification is QA-Scope because the seismic design of the ASIPs must not be compromised. No structural supports for the ASIPs will be affected by the modification. (SER 91-018)

84. MR 90-273 (Unit 2), Bus Tie Breaker's Control Cable Train Separation. MR 90-273 corrects electrical separation conflicts found in the control circuits for bus tie breakers 2B52-39C (2B01-2B03) and 2B52-26C (2B02-2B04). The specific electrical separation conflicts are that train "A" cables ZC2B39CC and ZC2B39CD are in the same raceways as train "B" cables ZD2B26CC and ZD2B26CD. The modification changes the existing control circuits for the bus tie breakers and the control circuits for the supply breakers to 2B01 and 2B02 to allow reclassification of the cables with electrical separation conflicts from safety-related to non-safety-related.

Summary of Safety Evaluation: During installation, the bus tie breakers will be racked out. The control power fuses for the 480 V supply breakers will be pulled prior to connection to the DC power source and the off time will be minimized. When the 480 V supply breaker control circuits are deenergized, the interlock with the 4160 V supply breaker will be disabled.

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

When the modification is complete, the control circuits for bus tie breakers 2B52-39C (train "A") and 2B52-26C (train "B") will have adequate electrical separation. While the modification does make changes to the control circuits for these breakers, it does not change the intended function of the affected portions of the control circuits (i.e., the interlock with the supply breakers 2B52-45B and 2B52-44B). The control circuits will remain functionally the same.

The safety-related function of these bus tie breakers is to trip upon either bus undervoltage or safety injection. MR 90-273 does not alter this function of the control circuits. The probability of an accident or equipment failure is not increased nor are the consequences of either of the two situations. (SER 91-089)

85. MR 91-001 (Unit 1) and 91-002 (Unit 2). EH Control System. The MR modifies the EH pump lockout circuit for each unit such that two independent low EH reservoir level signals are required to get the lockout. The change improves the reliability of the system by preventing a failure of a single level switch from locking out the EH pumps and tripping the unit.

Summary of Safety Evaluation: Spare contacts on an existing level switch which is currently used for alarms will be used to provide the redundant low level signal. The change requires minor rewiring inside the terminal box mounted on the EH skid, which will be done during a refueling outage. Testing includes manually actuating the level switches and verifying proper operation of the annunciator and lock-out.

The failure modes of the system remain the same except that now there are two level switches that could fail and disable the EH pump lockout. The consequences of his type of failure is that the EH pump could burn up if there were also a leak in the EH system. This has no impacts on safety related systems since the turbine stop valves and the governor valves would still close on loss of EH system pressure. This type of failure would be detected during annual testing and inspection of the switches.

The EH system is described in the "Turbine Control" portion of the FSAR starting in Section 10.2-4. The EH pumps are mentioned in this section but specifics about their control circuits are not discussed. The system description in the FSAR is not changed by this modification. The system still fails safe in that loss of voltage or EH fluid pressure will cause closure of the steam inlet valves. (SER 91-019)

86. MR 91-010*B (Common). 4160 V Electrical System. The MR replaces the existing four 10-pole test switches on 1(2)A05 and 1(2)A06. The new switches are compatible with the existing wiring and are seismically mounted.

Summary of Safety Evaluation: The presently-installed switches have plastic posts which are used to mount the switch covers. These plastic posts break when red locks are installed. One contact on each switch is used in the bus undervoltage protective circuit for testing. One contact on each bus is used in a jumper circuit around the 2A05 to 2A06 tie breaker auxiliary switch contact used in the emergency diesel supply breakers closing circuits.

New switches will be installed. The new switches have metal posts which are used to mount the covers. No new wiring or lugs will be necessary. Following installation of the new switches, continuity tests will be performed to verify that the connections are adequate and the switches are functional. No system functional logic changes are being made.

The new switches will be installed while safeguards buses 2A05 and 2A06 are out of service during U2R17. The buses will be deenergized one at a time for maintenance purposes. While a bus is out of service, its undervoltage protective circuits and emergency diesel supply breaker will not be needed. (SER 91-079)

87. MR 91-013 (Common). Structural. The modification adds a restraint to the FP header in the upper fuel oil pump house. The restraint prevents interaction between the FP header in the upper FOPH and the seismically qualified FP header in the lower FOPH. Thus, the lower FOPH will be protected from flooding concerns due to a rupture of the FP header.

Summary of Safety Evaluation: The lower fuel oil pump house (FOPH) contains safety-related fuel oil pumps used to supply the emergency diesel generators with

fuel oil in the event of a design basis accident with loss of offsite power. The current configuration of the fire protection (FP) header in the FOPH may lead to flooding in the lower FOPH due to a rupture of the header during a seismic event, thus, failing the fuel oil pumps. In order to alleviate this concern, the FP header in the upper FOPH will be structurally isolated from the lower FOPH by means of a restraint at the upper FOPH floor penetration. This restraint, along with a grouted penetration anchors the FP header at the floor penetration. The FP header below the floor penetration has been seismically qualified as-is. This MR anchors the fire protection header at the floor penetration to protect the lower FOPH FP header from the upper FOPH FP header effects.

The IWP addresses safeguards for having flammable material in a safe shutdown area. (SER 91-058)

88. MR 91-033 (Unit 2), Main Control Boards. MR 91-033 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 current operating philosophy is based on operation with rods out since an adequate negative reactivity insertion rate exists at the ARO condition. No TS or FSAR requirements were found regarding the upper withdrawal limit (bite limit) of the Bank D rods. 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 will continue to have a high alarm via the PPCS. Operators are notified of automatic rod motion by the clicking of the bank demand counters and the rod motion indication lamps. The new alarm will provide notification to the operator via a standard main control board alarm. This will alert the operator immediately to any unplanned reactivity changes.

The alarm will be 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 has been 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 will be mounted in the position left vacant by the removal of the RWBDH relay. The function of the RWBDH alarm will also be 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 will be installed during unit outages when the rod control system is deenergized. The alarm will be tested following installation by simulating an automatic rod motion signal.

The new alarm is expected to improve the operators awareness of any automatic rod motion. Any faults in the addition to the rod motion relay circuit will produce the same effects as faults in the rod motion indicating lamp circuits. These faults would produce alarms (rod control system non-urgent and urgent failure) which would clear as soon as automatic rod motion ceases. (SER 91-081)

89. MR 91-056 (Unit 1), Containment Spray System. Pipe whip restraint R101 on the Unit 1 spray line (RC-5201R-1) was modified to become a vertical pipe support during the IEB 79-14 program.

Summary of Safety Evaluation: The spray line was reanalyzed for pipe stress to account for thermal stratification effects of the type described in IEB 88-08. This reevaluation demonstrated that support R101 was developing large loads on the pressurizer spray nozzle. These loads were generated because the pressurizer needed to thermally expand over 2" in the vertical direction and R101 (which is approximately 5' removed from the spray nozzle) was restraining the spray piping from moving. To eliminate the excessive loads on the nozzle, it was decided to remove the support from the system even though the system was operable per the original thermal stratification analysis. A new analysis completed for this support configuration resulted in all pipe stresses and support loads satisfying all applicable code requirements.

In order to conform to the analyzed configuration, the shim stock for support R101 needs to be removed to allow thermal growth in the vertical direction. (SER 91-020)

90. MR 91-080 (Unit 1) and 91-081 (Unit 2), Residual Heat Removal System. The modifications replace bolted attachments and wall mounted attachments for the upper support braces of RHR heat exchangers 1HX-11A and 1HX-11B. The heat exchangers located in central primary auxiliary building at El. 8'; column lines N10 to N13.

Summary of Safety Evaluation: RHR heat exchangers 2HX-11A and 2HX-11B were qualified per PBNP operability criteria in NCR N-90-241, assuming the lower support legs of the heat exchangers can support the full load (i.e., the upper braces do not exist).

Existing expansion anchors, anchor base plates and brace connection bolts were replaced. The design of the new braces satisfies the requirements of the original plant design basis. (SER 91-014-00)

91. MR 91-163 (Unit 2), Main and Local Control Boards. MR 91-163 provides intercabinet structural support between miscellaneous relay rack 2C158 and engineered safeguards console 2C157 and new cabinet base anchorage for 2C158 and engineered safeguards consoles 2C157 and 2C156. These MRs allow for 2C158 to be upgraded from nonsafety-related to the QA and safety-related classification.

Summary of Safety Evaluation: The modification enhances the structural integrity of 2C158 and engineered safeguards consoles 2C157 and 2C156. These objectives are accomplished by: a) Installation of a new intercabinet structural brace between cabinet 2C158 and 2C157; and b) installation of new base anchorage for cabinet 2C158 and adjacent cabinets 2C157 and 2C156.

2C158 and engineered safeguards consoles 2C157 and 2C156 are located in the control building, cable spreading room on El. 26' between column lines F - D and 9.9 - 13.1.

2C158 and engineered safeguards consoles 2C157 and 2C156 will not be taken out of service or deenergized to perform the work. Although some of the equipment within these cabinets is important to safety, the work will be performed during the U2R17 refueling shutdown. TS 15.3.5 allows the engineered safeguards to be inoperable at this time. ICP 10.1, which is performed prior to the RSDs bypasses and blocks safeguards and AMSAC systems to prevent accidental actuation of these systems. ICP 10.1 will be in effect prior to the start of this modification and during its installation. ORT 3, ORT 6 and subsequent startup

testing, which are normally performed prior to reactor startup to verify proper operation of the safeguards and AMSAC systems, will be performed after completion of the modification. Precautions will be taken to prevent any metal filings or other debris from falling on or into equipment located within the affected cabinets. Therefore, operability of 2C158 and engineered safeguards consoles 2C157 and 2C156 and any equipment important to safety in the location of the upgrade will not be affected. (SER 91-073)

92. MR 91-211 (Unit 2), Main Steam System. MR 91-211 enhances the operating mechanism of the MSIVs to reduce the friction that opposes the closing force of these valves. The changes include using a different style of packing, using a non-corroding bushing, and using a deflector shield to prevent moisture from entering the air cylinder.

Summary of Safety Evaluation: Packing friction has proven to be a contributing factor to past valve failures to close. One of the changes replaces the outer packing rings with a new composite graphite material and reduces the total number of rings used on the valve shaft. The composite material provides 25%-50% less friction than the braided graphite rings currently in use. This type of packing has been 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.

Another change is a specific fix for 2MS-2017. The major cause of the valve's most recent failure was that the bushing on the air cylinder had rusted to the piston rod. The change replaces this steel bushing with a bronze bushing. Since the bronze is not as susceptible to corrosion, this should eliminate the possibility of a recurring failure. The bronze bushing is a suitable replacement for the steel bushing.

To minimize the potential for steam leaks, the carbon spacer in the stuffing box will be split, with half being put on each side of the packing. The previous configuration had the entire spacer inboard of the packing and any shaft deflection from the air cylinder holding the valve open which simply compressed the packing and provided a steam leak path underneath the shaft. Splitting the spacer minimizes the deflection of the shaft in the area of the packing and reduces the potential for future packing leaks. Also, using the composite packing allows the packing gland to be torqued higher, thereby providing a better seal without creating excessive packing friction.

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. This is done until either the packing loads meet manufacturer's recommendations or the valve fails to shut. If the valve fails before reaching the desired torque, the packing load will be reduced so the valve shuts. If the recommended packing loads cannot be attained in the cold condition, the packing will be retorqued and retested with the unit in the hot standby condition. This allows higher packing loads in the hot condition in order to more positively prevent steam leaks through the packing.

A precautionary measure installs a shield on the air cylinder and/or linkage to keep moisture out of the cylinder should a steam leak develop. The shield will be field-fit and will not interfere with valve operation. It will be made of light gauge material so it will not affect valve motion.

Post-installation testing will be performed to ensure that the changes do not interfere with valve stroke times (TS requirement of 5 seconds or less). (SER 91-098)

TEMPORARY MODIFICATIONS

1. TM 91-018 and NCR N-91-028. Ground Detection System. TM 91-018 addresses a corrective action identified in NCR N-91-028. The corrective action requires an interim ground fault detection circuit, prior to the installation of a permanent detection circuit. The 125 V ground detection consists of operation personnel monitoring the 125 V bus with a voltmeter once a shift.

   Summary of Safety Evaluation: This TM wires a jumper over the current limiting resistor which is in series with the alarming relay in the ground detection circuit on battery chargers D07, D08 and D09. This action increases the current to the relay and thus increases its sensitivity to grounding of the 125 V DC system.

   A more sensitive relay is installed, decreasing the required pickup current of the ground detection relay, thus increasing its sensitivity to grounding. The new more sensitive relay has a higher coil resistance which offsets part of the increased sensitivity provided by the new relay.

   The new relay with its 5000 ohm coil provides a conservatively large enough resistance to limit the short circuit current to be under the minimum BFD relay dropout current of 11.7 ma.

   By installing a new more sensitive relay and jumping over a limiting resistor, the ground detection circuit is designed to detect grounds as high as 2500 ohms at a battery voltage of 134 V.

   The failure mechanism of the new relay is the same as the existing circuit with the resistor/relay combination. Therefore, the new circuit does not adversely affect the DC system. (SER 91-036)

2. TM 91-023. 1RH-742 Stem Clamp. The TM clamps valve 1RH-742 shut during power operation.

   Summary of Safety Evaluation: 1RH-742 is Seismic Class I. By inspection, the handwheel weighs much more than the clamp. The handwheel will not be installed when the clamp is in place. Therefore, when the clamp is installed, the ability of 1RH-742 and adjacent piping and supports to withstand a seismic event will not be diminished.

   1RH-742 is a containment isolation valve. It is safety-related because it forms part of the RHR system pressure boundary. For these reasons, seat tightness is important during power operation. To ensure that the clamp secures the valve fully shut, 1RH-742 with the clamp installed will be tested for seat leakage per IT-530 or equivalent prior to power operation.

   Valve 1RH-742 is used to return water to the RWST from the refueling cavity via the RHR system. The valve is not directed to be opened by any emergency procedures. If some plant condition requires the valve to be opened, this can be done by removing the clamp, screwing on the handwheel, and opening the valve in the normal manner.

Although the valve is within the scope of the QA program, non-QA parts were used for the clamp because QA-threaded rod was not available. This is acceptable because the clamp will be tested at its design load. (SER 91-043)

3. TM 91-027, Temporary Lead Shielding of Valves 1PCV-431A and 1PCV-431B. Temporary lead shielding is required around Unit 1 spray line control valves 1PCV-431A and 1PCV-431B to reduce exposure to workers repairing pipe support RC-2501R-1-R101 located ~15" from either valve.

Summary of Safety Evaluation: The lead shielding was evaluated for its effect on the spray line piping and found to be acceptable. Pipe stresses, support loads and equipment loads are below applicable Code allowables. The analysis is for deadweight effects on the reactor coolant system only. The addition of the lead blankets may affect the seismic operability of the reactor coolant system but this is not a concern while the reactor is in the cold shutdown condition during a refueling outage. The lead shielding will be in place for less than one week, and will only be utilized during the U1R18 refueling outage while the reactor is in the cold shutdown condition. (SER 91-045)

4. TM 91-036, Electrical Distribution. TM 91-036 directs that the leads to H52-30 trip device from X03 and bus Section 5 lockouts be lifted and potentially damaged cable be replaced during a Unit 2 outage. Cable damage is indicated by breaker indication at the system control center flickering on and off, and voltage checks of the wiring indicating partial voltage.

Summary of Safety Evaluation: Circuit breakers H52-30 and H52-06 are in series supplying 13.8 kV to bus H03 from transformer 2X03. H03 supplies low voltage station auxiliary transformer 2X04, which supplies safeguards buses 2A05 and 2A06 via 2A03 and 2A04.

Breaker H52-30 trips of concern are: Trip on X03 lockou[illegible] on bus Section 5 lockout and trip on breaker H52-06 tripping from auxiliary [illegible]act on H52-06. The first two trips will be disabled.

Breaker H52-06 trips are: Trip on X03 lockout and trip on bus Section 5 lockout.

X03 and bus Section 5 lockout trips to H52-30 will be defeated. These trips are duplicated on series breaker H52-06. The combination of these two breakers will continue to function as before the TM. The defeated trips result in a reduction in redundancy in this circuit. This reduction in redundancy does not decrease the reliability of the AC system since the 13.8 kV system is not single failure proof. The AC system will continue to perform its safety function. (SER 91-064)

5. TM 91-049: Main Steam. The TM repairs a stuffing box to body leak on the south side of 2MS-2017A, main steam nonreturn valve. The OD of the stuffing box flange will be drilled and tapped to enable installation of injection valves and subsequent injection of sealant compound. This may also require peening of the flange OD at the mating surface to the valve body to contain sealant. After sealant is injected, the counterweight will be deflected to verify freedom of movement of the valve shaft.

Summary of Safety Evaluation: Leak repairing of the stuffing box flange area on a non-return valve involves drilling holes in the stuffing box flange and injecting sealant via injection adapters. Calculation P-88-023 addresses bolt stresses as a result of changing the gasket design to a full-face gasket, thus increasing the stud loading. Stresses at system design pressure are acceptable. Stresses seen during

sealant injection are above Code allowable (25,000 psi) but not considered above those values that could be seen during stud torquing.

Sealant injection will not impair the disc/arm stroking capability because the sealant does not directly touch the valve shaft. The freedom of movement of the valve shaft can be verified by deflecting the shaft toward the closed position by applying a force on the counterweight arm and then allowing it to return. The deflection will be slight and will not affect flow rates. (SER 91-105)

MISCELLANEOUS EVALUATIONS

1. ECR PB-91-025, Rod Speed Auto Control Circuit. During post-installation testing for MR 88-018*F, the maximum rod speed specified could not be achieved within tolerance. The maximum rod speed that could be obtained was ~68 steps per minute. The ideal maximum rod speed is 72 steps per minute and the test has a tolerance of ±2 steps per minute. Investigation showed that with the new, lower resistance meter, the voltage necessary to obtain maximum speed could not be developed in the Foxboro controller (1TM-401DD). ECR PB-91-025 replaces the existing 190 ohm resistor, located in 1TM-401DD, with a 195.5 ohm resistor. Following installation of the 195.5 ohm resistor, the maximum rod speed should be ~72 steps per minute.

   Summary of Safety Evaluation: The higher resistance will result in a higher voltage across the meter and rod speed control rack. This higher voltage will lead to higher rod speeds. The maximum rod speed should be 72 steps per minute based on the Setpoint Document. Based on calculations, the maximum and intermediate rod speeds should remain within tolerance with the new resistor installed. (SER 91-040)

2. MWR 903789, Spent Fuel Pool Bridge. MWR 903789 mounts a rack on the handrail of the spent fuel pool (SFP) bridge, for containing needed video equipment used during inspections of the SFP.

   Summary of Safety Evaluation: The addition of the rack does not affect the seismic rating of the bridge, or overload the power supply of the bridge. The rack is mounted above the bridge, and in an area of low usage. The rack does not interfere with any fuel handling procedures or hinder the bridge's performance. The SFP bridge is Seismic Class 2, and is designed such that it will not fall during an SSE, and it will also not drop a fuel assembly. The rack will not affect this design feature of the bridge. Total weight of the rack and equipment is less than 200 lbs. If this fell into the SFP, it would not cause significant fuel damage due to the light weight, buoyancy effect of water, fuel racks protecting stored fuel and stored fuel pins being protected by top nozzle, and the inability to gain significant sideways movement to strike a hanging fuel assembly. The rack will most likely fall onto the walkway of the SFP bridge. (SER 91-060)

3. MWRs 904983, 904983, 913875, 913876 and Associated SMPs: Service Water. The work includes the replacement of piping downstream of SW-48 and SW-50 and upstream of SW-49 for G-01 and piping downstream of SW-51 and SW-56 and upstream of SW-58 for G-02. Replacement will be with carbon steel piping equivalent in weight.

   Summary of Safety Evaluation: The service water supply and return piping to G-01 and G-02 emergency diesel generators is to be replaced because of unacceptable pipe wall ID pitting. This includes piping downstream of SW-48 and SW-50 and upstream of SW-49 for G-01 and piping downstream of SW-51 and SW-56 and

upstream of SW-58 for G-02. During the supply piping replacements, the root header isolation valves, SW-47 from the south header when G-01 is done and SW-65 from the north header when G-02 is done, will be shut to allow hydrolancing of piping that is accessible when the supply piping is cut off. Although the redundant supply to the operable diesel will be red tagged shut, the piping will be in a configuration which can be utilized, if an emergency arises and the redundant supply is needed; i.e., there will be no unisolable holes in the service water headers at the point of connection to the piping being replaced. The piping was evaluated for seismicity for that time when the pitted pipe is removed. Deadweight support at the valves adjacent to the removed piping will be added per the work procedure but there is no effect on the seismicity of the remaining piping.

Some of the flushing requires removing the fire suppression system from service. Fire watch requirements and a need for an alternate fire suppression method will be requirements of the work procedures. Although the diesel whose service water piping is being replaced will be out of service, the fire protection in the respective room remains in service, or compensatory action taken, since the normal and alternate (Appendix R) power supplies to several service water pumps go through these rooms (P-32B and P-32F in G-01 room, P-32C and P-32E in G-02 room). (SER 91-071)

4. MWRs 913229 and 913230, Buildings & Structures. The work replaces the control room door hinges, and aligns and adjusts control room doors 61 and 65. The work requires removal of the doors for a short period of time (~4 hours).

Summary of Safety Evaluation: The doors are for security and fire protection. A security officer and a fire watch will provide compensatory measures to assure fire protection and security requirements are maintained. The hinge meets the required standards for the door.

The door is part of the control room environmental envelope. The capacity of the air conditioning system will not be affected by the removal of the door. The room temperature in the immediate vicinity of the door may increase slightly but not enough to affect the operation of any equipment. There will be roughly 700 cfm of air exhausting through the door opening. This is air normally lost by exfiltration from the control room. This exhaust air will act as an air curtain to limit the movement of warm turbine building air into the control room.

The door does not provide radiological protection. Movable lead shielding is provided and use of the shielding will not be affected. The door provides a barrier for control room pressurization. During accident conditions, the control room is to be kept at 1/8" water gauge pressure greater than the surrounding spaces. This prevents the infiltration of contamination into the control room. Control room pressurization can be obtained by taping a plastic tarp over the opening. If necessary, the tarp can be backed by a sheet of fire-rated plywood or a metal plate. There is no time constraint for the room pressurization. The radiological dose calculation is based on 30 days. In addition, there will normally be a time lag between an accident and a radiological release that would require control room pressurization. It may be acceptable to continue with the reinstallation of the door before establishing room pressurization.

FSAR Appendix E addresses high energy pipe failures. Section 3.2.3 indicates the pressure from a main steam line break would be 0.258 psi for 7.55 seconds at the window next to the door. Per Section 2.0.5.6 of the Point Beach Initiation Event notebook, the probability of a main steam line break outside of containment, excluding spurious operation of the safety valves, is 2.5E-4 per year. The probability of a steam line break during the roughly 4 hours that the door is out of the opening is 1.14×10^{-7}, [(2.5E-4/year)/(8760 hr/year) x 4 hours]. This is less than 1×10^{-6}. A high energy pipe line break is not a credible event. The movable lead shielding could be located between the door opening and the main steam lines to provide shielding from a steam line break if desired.

The door being removed is similar in impact to the control room cleanup HVAC being out of service. The control room cleanup HVAC has a 7-day LCO associated with its performance. Thus, repair of the door resulting in the degradation of maintaining a positive pressure in the control room for one shift is not an unreviewed safety question. (SER 91-065)

5. MWR 914961: TS Table 15.3.5-2 requires a minimum of three operable power range channels with a minimum degree of redundancy of two. Removing instrument power fuses accommodates the redundancy requirement by placing the hi-flux hi trip signal from channel N44 in the tripped condition. To facilitate troubleshooting and calibration, channel N44 is to remain inoperable by disconnecting the detector signal cables or reducing indicated power to less than actual power with instrument power fuses installed. To accommodate this condition, the hi-flux hi-trip signal will be generated by placing selector switch S1 in Position 11 and test switch S5 in test in both trains of reactor protection logic. Maintaining channel N44 trip logic matrix relays in trip ensures the minimum degree of redundancy is maintained while N44 is inoperable.

Summary of Safety Evaluation: Powering up channel N44 while having it in a condition such that indicated power is less than actual power for troubleshooting or calibration purposes disables channel N44 permissive logic relays for P7, P8, P9, and P10. This results in the permissive logic being reduced to 2/3 as opposed to a normal logic of 2/4.

The permissive function is when reactor power is greater than the permissive setpoint. 2/4 permissive logic signals are required for P7, P8 and P9 to unblock their respective reactor trip signals, while 2/4 permissive logic signals are required for P10 to block the NI reactor trips identified above. Therefore, 3/4 permissive logic signals are required to block the reactor trips associated with P7, P8 and P9 and to unblock the reactor trip signals associated with P10. Powering up channel N44 with the input signals disabled or creating a condition where indicated power is less than actual power, while varying test signals, could result in a varying permissive logic status for channel N44.

The remaining three power range channels will remain in service and operable, ensuring that the permissive circuits are maintained in the condition such that the reactor trip signals are not being blocked by the permissive when power is above the permissive setpoint. ICP 10.2 ensures the P10 unblock feature from channel N44 is maintained if actual power decreases below the P10 setpoint. Failure of one of the remaining three power range channels while N44 is inoperable would still not inhibit any reactor trip signals via the permissive circuits because 2/4 power range channels are available to retain the permissive in the unblocked condition.

Declaring channel N44 inoperable will not place an additional burden on any reactor protection circuits or degrade the reactor protection system. (SER 91-100)

6. WEP 91-102, Unit 1 & 2 Steam Generator Fatigue Evaluation Summary Report, Secondary Side Pressure Testing and Hydros. The evaluation discusses the effect of the ASME Section XI-required and PBNP requested increase in the number of hydrostatic tests and leak tests and the conditions under which they are performed resulting from steam generator technical manual inconsistencies.

Hydrostatic tests are performed to verify pressure boundary integrity following modifications or repairs, or to address Code requirements. Leak tests are performed at a lower than operating pressure and are done to verify the absence of leaks following plugging, sleeving or annual operational cycles.

As a result of discrepancies in the test parameters for the steam generator leak tests and hydro tests as stated in the Technical Specifications, SG vessel design/stress report, SG equipment specification and the technical manuals, these parameters required clarification along with their associated fatigue factors.

Summary of Safety Evaluation: The increase in the number of hydro/pressure tests on the secondary side of the SGs along with the test temperature parameters of ASME/Appendix B, do not involve an unreviewed safety question.

The operation of the SGs following the parameter changes presented by the manufacturer regarding hydro/leak tests will not have an adverse effect on the pressure boundary integrity of the SGs and do not represent an unreviewed safety question. There have been no physical changes or alterations to the steam generators. (SER 91-026)

7. U1C19 Reload Core. The Unit 1 Cycle 19 reload contains 12 fresh Region 21A upgraded Optimized Fuel Assemblies (OFA) at 3.6 w/o, 16 fresh Region 21B upgraded OFA at 4.0 w/o, 16 Region 20A upgraded OFA, 12 Region 20B upgraded OFA, 1 Region 20C upgraded OFA, 12 Region 19A upgraded OFA, 15 Region 19B upgraded OFA, 12 Region 18A OFA, 16 Region 18B OFA, 8 Region 17B OFA, and 1 Region 9 Unit 2 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 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 9700 and 11000 MWD/MTU. Actual U1C18 burnup was approximately 10750 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 approximately 0 to 10 ppm of residual boron at the Cycle 19 rated power condition of 1518.5 MWt) plus 1500 MWD/MTU power coastdown; and c) there is adherence to the plant operating limitations given in the Technical Specifications. (SER 91-032)

8. TS 15.3.10.D, DCS 3.1.5, Basis for Continued Operation Unit 1 RPI K7 and C7. This basis for continued operation is required to justify that the C7 and K7 RPIs are operable and the increased monitoring specified in TS 15.3.10.D.3 is not required when the RPIs are responding normally.

Summary of Safety Evaluation: Rod position indicators K7 and C7 in Bank D have behaved erratically for a number of years. MWR history shows instances of this occurring back to May 1988. Data trended with the PPCS shows that the erratic behavior may occur on a length of the RPI coil stack between 125" and 135" or 195 and 211 steps with reduced RCS temperatures. The DCS Handbook (DCS 3.1.5) provides guidance as to declaring RPIs out of service. Using the

historical data gathered on these RPIs, and the DCS Handbook guidance, control operators are able to determine the operability of the RPIs. The RPIs have generally behaved normally; that operators operators are able to recognize problems with the RPIs. Therefore, there is a basis for not declaring the RPIs out of service continuously. These RPIs should be declared inoperable when the criteria established by DCS 3.1.5 are met. Increased surveillance is required when the RPIs are inoperable.

The evaluation determined that the C7 and K7 RPIs are normally operable. Allowing these RPIs to be used for rod position indication does not contribute to the consequence of an accident or pose an unreviewed safety question. Continued operation with these RPIs is considered acceptable. (SER 91-031)

9. MISC OPS, Use of Assembly U07 in U1C19 With Damaged Upper Spring Clip Grid.

   Summary of Safety Evaluation: Safety evaluations for the control rod guide tube flexure and split pin failures address the concerns for loose parts in the reactor coolant system. The potential for loose parts from the U07 fuel assembly damaged grid is considered to be less significant than the split pin parts, which were previously shown to not constitute an unreviewed safety question. The potential for damage caused by flow maldistribution from the damaged grid has been judged to be minimal. The field anomaly report for fuel assembly U07 states that based on past experience with similar grid damage, it is recommended that the assembly is reused as is with handling precautions. Therefore, the use of fuel assembly U07 with a damage upper grid does not constitute an unreviewed safety question for Unit 1 Cycle 19. (SER 91-032-01)

10. Justifications for Continued Operation: JCOs for valves having Limitorque motor operator torque switches made of melamine material. Torque switches of the "cam" design used in Limitorque motor operators Model SMB 00/000 made of melamine material have been found to not perform as designed. The melamine material is susceptible to post-mold shrinkage causing the moving parts of the torque switch to bind resulting in the torque switch tripping at lower stem thrust/torque levels.

    Limitorque issued a 10 CFR Part 21 notification dated November 3, 1988, on this nonconservative failure. PBNP has experienced isolated cases of this type of torque switch binding, however, this is not a widespread problem and PBNP has had many successful years of operation with the melamine torque switches.

    a. AF-4020, P38B AF Pump Discharge to 2HX-1B Steam Generator.

       Summary of Safety Evaluation: The AF-4020 valve operator is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

       AF-4020 is a safety-related component. The valve is normally in the open position. The primary function of AF-4020 is to shut to isolate the Unit 2 "B" steam generator in the event of an auxiliary feedwater pump actuation (P-38B) for Unit 1.

AF-4020 can receive either an automatic open signal or an automatic close signal depending on the plant conditions. In the event that AF-4020 did not function properly due to a torque switch failure, feedwater would still be available since there is a second electric auxiliary feedwater pump system and a steam-driven auxiliary feedwater pump system.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem thrust for proper valve closure. Proper torque switch operation was documented during 1987 preventive maintenance. The valves are verified to stroke properly monthly per IT-10. (SER 91-039)

b. SW-2817, HX Water Treatment Area Cooling Coil Inlet.

Summary of Safety Evaluation: SW-2817 is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2817 is a safety-related component. The valve receives an automatic closure signal upon the receipt of either a Unit 1 or Unit 2 safety injection (SI) signal without enough service water pumps starting.

SW-2817 is used to isolated the service water supply to water treatment. The valve is shut to isolate service water for water treatment maintenance or would automatically shut to isolate non-essential service water during an SI actuation requiring isolation of non-essential service water.

In the event that SW-2817 did not function properly due to torque switch failure, the service water system would not be placed in a condition outside of the design basis. FSAR Section 9.6.2 states that the service water system was designed to provide sufficient service water flow in case one of the motor-operated valves for isolation of nonessential services fails to close.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem thrust for proper valve closure. Proper valve closure under full-flow differential pressure is performed quarterly by IT-72. The valve also shuts automatically during performance of ORT 3 (Loss of AC Test) for Unit 1 and Unit 2. SW-2930A also has a melamine torque switch installed. Since the valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-039-01)

c. SW-2869, North Service Water Header to West Service Water Header.

Summary of Safety Evaluation: SW-2869 operator is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2869 is a safety-related component. The valve does not receive any automatic closure or opening signals.

The service water supply system is configured as a ring header that consists of a south header, west header and a north header. In the event of a SW pipe break, the affected header can be isolated and the service water supply to equipment important to safety can be realigned to the unaffected header. SW-2869 is the remotely-operated valve that provides header isolation between the north header and the west header.

In the event that the north service water header requires isolation and the SW-2869 valve operator does not function properly due to a torque switch failure, the south and west header can be remotely isolated using the south header to west header isolation valve (SW-2870) or the manual valve installed adjacent to SW-2869. AOP-9A provides guidance for the operators to manually isolate the leaking portion of the service water header.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem torque for proper valve closure. Proper valve operation is checked quarterly by IT-07. SW-2870, SW-2890 and SW-2891 also have melamine torque switches installed, since the valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-039-02)

d. SW-2870, South Service Water Header to West Service Water Header.

Summary of Safety Evaluation: SW-2870 is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2870 is a safety-related component. The valve does not receive any automatic closure or opening signals.

The service water supply system is configured as a ring header that consists of a south header, west header and a north header. In the event of a SW pipe break the affected header can be isolated and the service water supply to equipment important to safety can be realigned to the unaffected header. SW-2870 is the remotely operated valve that provides header isolation between the South header and the West header.

In the event that the south service water header requires isolation and the SW-2870 valve operator does not function properly due to a torque switch failure, the south and west headers can be remotely isolated using the north header to west header isolation valve SW-2869 or the manual valve installed adjacent to SW-2870. AOP-9A provides guidance for the operators to manually isolate the leaking portion of the service water header.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem torque for proper valve closure. Proper valve operation is checked quarterly by IT-07. SW-2869, SW-2890 and SW-2891 also have melamine torque switches installed, since the

valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-039-03)

e. SW-2890, North Service Water Header to South Service Water Header Crossconnect.

Summary of Safety Evaluation: SW-2890 is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2890 is a safety-related component. The valve does not receive an automatic closure or opening signal.

The service water supply system is configured as a ring header that consists of a south header, west header and a north header. In the event of a SW pipe break, the affected header can be isolated and the service water supply to equipment important to safety can be realigned to the unaffected header. SW-2890 is one of two valves (SW-2891 is the other) located in the pumphouse used to isolate the discharge of the P-32A, P-32B, and P-32C pump discharges (south header) from the P-32D, P-32E, and the P-32F pump discharges (north header).

In the event the SW-2890 operator did not function properly due to a torque switch failure, the SW-2891 valve operator could be used for isolating the south header from the north header. The ability to isolate the south header from the north header using manual valves is also possible.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem torque for proper valve closure. Proper valve operation is checked quarterly by IT-07. SW-2870, SW-2869 and SW-2891 also have melamine torque switches installed, since the valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-039-04)

f. SW-2891, South Service Water Header to North Service Water Header Crossconnect.

Summary of Safety Evaluation: SW-2891 is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2891 is a safety-related component. The valve does not receive an automatic closure or opening signal.

The service water supply system is configured as a ring header that consists of a south header, west header and a north header. In the event of a SW pipe

break, the affected header can be isolated and the service water supply to equipment important to safety can be realigned to the unaffected header. SW-2891 is one of the two valves (SW-2890 is the other valve) located in the pump house used to isolate the discharge of the P-32A, P-32B, and P-32C pump discharges (south header) from the P-32D, P-32E, and the P-32F pump discharges (north header).

In the event the SW-2891 operator did not function properly due to a torque switch failure, the SW-2890 valve operator could be used to isolate the south header from the north header. The ability to isolate the south header from the north header using manual valves is also possible.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem torque for proper valve closure. Proper valve operation is checked quarterly by IT-07. SW-2869, SW-2870 and SW-2890 also have melamine torque switches installed, since the valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-093-05)

g. SW-2930B, HX-13B Spent Fuel Pool Heat Exchanger Outlet.

Summary of Safety Evaluation: SW-2930B is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

SW-2930B operator is a safety-related component. The valve receives an automatic closure signal upon the receipt of either a Unit 1 or Unit 2 safety injection (SI) signal without enough service water pumps starting.

SW-2930B is used to isolate the service water supply to the "B" spent fuel pool heat exchanger. The valve would be shut to isolate service water to the "B" spent fuel pool heat exchanger or would automatically shut to isolate nonessential service water during an SI actuation requiring isolation of nonessential service water.

In the event that SW-2930B did not function properly due to torque switch failure, the service water system would not be placed in a condition outside of the design basis. FSAR Section 9.6.2 states that the service water system was designed to provide sufficient service water flow in case one of the motor-operated valves for isolation of nonessential services fails to close.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem thrust for proper valve closure. Proper valve closure under full flow differential pressure is performed quarterly by IT-72. The valve also shuts automatically during the performance of ORT 3 (Loss of AC Test) for Unit 1 and Unit 2. SW-2817 also has a melamine torque switch installed, since the valves are periodically tested and the torque switch failure is age and temperature dependent, it is reasonable to expect only one torque switch to fail at one time. (SER 91-039-06)

h. 2CC-754A, P1A Reactor Coolant Pump Component Cooling Inlet MOV.

Summary of Safety Evaluation: The 2CC-754A operator is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by 12/31/92.

2CC-754A is a safety-related component. The valve is used as a containment boundary isolation valve. The valve does not receive an automatic closure signal, operator action is required for valve stroking.

2CC-754A is used to isolate the component cooling water to the "A" reactor coolant pump (RCP). The valve would be shut in the event of a component cooling water line break or in the event of a rupture of the "A" RCP thermal barrier.

In the event that the 2CC-754A operator did not function properly due to a torque switch failure, the 2CC-719 valve operator could be used to isolate component cooling water to containment.

The failure mode of the melamine torque switch will cause the valve operator to torque-out prematurely and not provide the required stem thrust for proper valve closure. Proper torque switch operation was verified and documented during 1987 preventive maintenance. Proper valve closure to provide a tight seal is verified every refueling outage through the performance of ORT 68. (SER 91-039-07)

i. 2SI-826B, P15A/B Safety Injection Pump Suction from Boric Acid Storage Tank Parallel Isolation.

Summary of Safety Evaluation: The 2SI-826B valve operator is presently operating properly. The valve operator is not located in an environment which exposes the torque switch to continuous temperatures near or exceeding 120°F. Considering plant experience, operator temperature environments, and that the melamine shrinkage is time and temperature dependent, it is reasonable to expect that the torque switch will not fail prior to its scheduled preventive maintenance to be performed by December 31, 1992.

The 2SI-826B valve is a safety related component. The valve receives an automatic opening signal upon receipt of a Unit 2 safety injection (SI) signal. It provides suction from the boric acid storage tanks to the safety injection pumps. Valve 2SI-826C provides opposite train redundancy for 2SI-826B.

In the event that both 2SI-826B and 2SI-826C fail to reach intermediate valve position within 1.5 seconds, then the 2SI-825A and 2SI-825B valve operators receive an open signal. The 2SI-825A and 2SI-825B valve opening provides suction from the refueling water storage tank (RWST) to the suction of the SI pumps.

The 2SI-826B valve are required to shut once the boric acid storage tank reaches a low low level. 2SI-826A could be used to isolate the boric acid storage tank in the event that 2SI-826B failed to shut.

The failure mode of the melamine torque switch cause the valve operator to torque-out prematurely and not provide the required stem thrust for proper valve closure. Proper torque switch operation was verified and documented during 1987 preventive maintenance. Proper valve closure is monitored continuously by the fact that the RWST is not diluting the boric acid storage tanks. The 2SI-826B is verified to stroke properly monthly per IT-02. (SER 91-039-08)

11. U2C18 Reload. The Unit 2 Cycle 18 reload contains 16 fresh Region 20A upgraded optimized fuel assemblies (OFAs) at 3.6 w/o, 12 fresh Region 20B upgraded OFAs at 4.0 w/o, 12 Region 19A upgraded OFAs, 16 Region 19B upgraded OFAs, 12 Region 18A upgraded OFAs, 16 Region 18B upgraded OFAs, 12 Region 17A OFAs, 16 Region 17B OFAs, 8 Region 16B OFAs, and one Region 8 Unit 1 standard fuel assembly. The Cycle 18 core is the third reload containing a full region of upgraded OFAs fuel for Unit 2. Upgraded OFAs fuel is the subject of TS Change Request 127, which has been approved by the NRC.

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 18 evaluation, it is concluded that the Cycle 18 design does not cause previously acceptable safety limits to be exceeded, provided that: a) Cycle 17 burnup is bounded by 10500 and 11050 MWD/MTU. Actual U2C17 burnup was approximately 10778 MWD/MTU; b) Cycle 18 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 approximately 0 to 10 ppm of residual boron at the Cycle 18 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. (SER 91-092)

12. Operation with the Rod Drop Signal from One Channel of NIS in Bypass: The rod drop signal from power range nuclear instrumentation channel 1N44 will be placed in bypass for testing. Voltage spikes on 1N44 resulted in two turbine runbacks and the channel must be tested to identify the cause.

Train "A" of the signal produces a turbine runback via load reference at a rate of 200% per minute. Train "B" of the signal produces a turbine runback via load limit to 80% of full power and blocks automatic rod withdrawal. Bypassing the rod drop signal from 1N44 changes the logic for each of these functions from 1/4 to 1/3 and increases the probability that a dropped rod in the core quadrant near the detector for channel 1N44 may not be detected by the nuclear instrumentation system.

Summary of Safety Evaluation: The rod drop signal from power range nuclear instrumentation channel 1N44 will be placed in bypass until the spiking problem with NIS channel N44 is resolved. Voltage spikes on 1N44 have resulted in two turbine runbacks recently, and the channel must be tested to identify the cause.

Train "A" of the signal produces a turbine runback via load reference at a rate of 200% per minute. Train B of the signal produces a turbine runback via load limit to 80% of full power and blocks automatic rod withdrawal. Bypassing the rod drop signal from 1N44 changes the logic for each of these functions from 1/4 to 1/3 and increases the probability that a dropped rod in the core quadrant near the detector for channel 1N44 may not be detected by the nuclear instrumentation system.

Even if the dropped rod is not detected by any of the other three detectors, the rod bottom signal from the rod position indication system would identify a dropped rod and initiate protective actions similar to those initiated by the nuclear instrumentation system. The rod position indication system is identified in the

Technical Specifications as a redundant means of detecting a dropped rod providing justification for bypassing the rod drop signal from one channel. (SER 91-086)

13. Reactor Coolant System Foreign Material: During the Unit 2 Refueling 17 outage a foreign object was identified during the closeout inspection of the "A" steam generator. During subsequent retrieval efforts and visual evaluation this object was determined to be "hard sludge." The safety evaluation analyzes the existence of loose, hard sludge in the secondary side of the steam generators and its existence as related to continued safe operation of the plant. The incident involves the decision to not remove a piece of what was determined to be "hard sludge" after retrieval efforts failed to dislodge the piece from between two tubes during the U2R17 refueling outage.

Summary of Safety Evaluation: The foreign object was sized at approximately 5/16" thick by approximately 3/8" wide by approximately 1¼" long. It appears to be propped up under a pile of hardened sludge with its front end sticking into and resting on the hot leg annulus approximately 3/8" out from the outermost row of the steam generator.

The steam generators have operated with this hard sludge accumulation since they were initially placed in operation. The hard sludge broken loose during maintenance operations that is not removed accumulates in the low flow (kidney) regions once the steam generators are returned to operation. It is assumed that if the flow is sufficient to erode and move the object, it will return to a low flow area once again. Wear to the tubes a foreign is adjacent to or in contact with is also a consideration. The eddy current history of the Unit 2 steam generators has shown a 5% per year growth rate associated with the thinning phenomenon. The tubes in contact with this object would be expected to experience the same if the object is the same consistency as the other known hard sludge. This has been proven by virtue of the removal process breaking off and grooving the object during gripping attempts and by the fact that eddy current examinations have detected no change in permeability, conductivity, or geometry in the tubes in this area. Further eddy current examinations will be performed at each refueling outage to ensure this situation does not change and cleanliness inspections following sludge lancing are part of the base scope of sludge lancing. (SER 91-097)

SPARE PARTS EQUIVALENCY EVALUATION DOCUMENTS (SPEEDs)

1. SPEED 90-085, Limit Switches. NAMCO Controls issued Revision R to the environmentally qualified (EQ) EA180 series limit switches. These new switches were qualified by NAMCO Report No. QTR-155, Revision 0. The environmental conditions that the new switches were qualified to bound the conditions of NAMCO Report No. QTR-105. The SPEED allows for use of these new switches in EQ applications.

Summary of Safety Evaluation: NAMCO Model EA180-15302 limit switches are used on various air operated valves (AOVs) for position indication only. This position indication is to verify that the associated AOV has closed following a containment isolation after a LOCA or HELB accident. As a result, the use of the new switches will not increase the probability or consequences of an accident.

The new limit switches have different contact block and contact carrier material than the switches we currently use. The new material is a glass-filled rather than an asbestos-filled phenolic thermoset plastic. Report No. QTR-155 shows that the glass-filled material is as good or better than the asbestos-filled. The report also shows that the new switches meet the same physical performance and environmental requirements as the switches we currently use. (SER 91-006)

2. SPEED 91-045, Substitution of Rockbestos Wire for General Electric SI53917. The SPEED documents the use of Rockbestos firewall SR wire as a direct replacement for existing GE SI53917 on any system, component or structure.

Summary of Safety Evaluation: Both wires are constructed of the same materials and wire gauge. Both have the same voltage ratings. Both meet the criteria of the IEEE 383-1975 flame test. In addition, the Rockbestos wire meets the criteria of IEEE-323 (Class 1E nuclear qualification). The Rockbestos wire uses a conductor made up of 65 strands versus 19 strands for the GE wire. This improves the wire's flexibility for easier installation. (SER 91-054)

3. SPEED 91-048, Plug Upgrade in Copes-Vulcan Valves. The trim on Copes-Vulcan Type D-100 valves was upgraded by the manufacturer from a wear tip trim to a cascade trim. This was done to reduce the valve trim erosion which resulted from cavitation across the seating surface. The cavitation was caused by a large pressure drop along the smooth valve seat. The cascade trim has a series of concentric grooves machined into the plug. These grooves reduce the pressure in steps which reduces vapor formation.

Summary of Safety Evaluation: MS-2083 and MS-2084 are both containment isolation valves for the steam generator sample lines. According to the manufacturer, the replacement trip assembly weight is approximately equal to the original trim and the replacement will not affect the seismic qualification of the manufacturer's valves. Valve leakage will not be a problem because the valve seat was redesigned to fit the cascade plug. Potential leakage will be monitored because an inservice test is annually performed on the valves. (SER 91-055)

4. SPEED 91-066, Nitrogen System Check Valve Replacement. SPEED 91-066 changes out check valves NG-1675, 1676, and 1677. These check valves are designed to isolate the nitrogen bubblers from the nitrogen header in case the nitrogen header loses pressure. The existing valves are lift check valves intended for liquid service. The replacement valve will be a poppet-style check valve that is better suited for this application.

Summary of Safety Evaluation: The valves are used in the boric acid tanks bubbler system to isolate the nitrogen header from the backup nitrogen bottles. This is an ~70 psi working pressure system. The installed valves are Whitey 3/8" stainless steel lift check valves with Swagelok tube connections. The replacement valves are Nupro "CH" series 3/8" stainless steel poppet-style check valves with Swagelok tube connections. The lift check valves are designed for gas and liquid service but are best utilized in high pressure liquid systems. The poppet-style check valve is also designed for gas and liquid service, but is better utilized in low pressure gas systems like the nitrogen system. The poppet-style check valve has a 1/3 psi cracking pressure and soft seats for superior seating characteristics than the metal-to-metal seat of the lift check valve. The poppet-style valve is equivalent to the lift check valve in material, pressure and temperature ratings, and tube connections. The valves differ in style (lift check versus poppet style), and the size of flow orifices and flow characteristics. But these differences are what make the poppet-style valve better suited for this application. (SER 91-088)

5. SPEED 91-070, Replacement of Rod Insertion Limit Computers. The change replaces the Foxboro Model 660 rod insertion limit computers with a Westinghouse programmable computing unit Model TMD 9000-08/00/00/00/00-08-08-03.

Summary of Safety Evaluation: Due to the discontinuation of the Foxboro Model 660 computing unit, it is necessary to replace the rod insertion limit computers. Westinghouse, using the Foxboro specifications, has developed an equivalent unit that meets or exceeds the original Foxboro specifications. The Westinghouse Model TMD 9000-08/00/00/00/00-08-08-03 was tested and was found to have better linearity and repeatability. Because of the design of the Westinghouse high limiter in the computing unit, the component will be more reliable and have less of a tendency to drift than the Foxboro unit.

The rod insertion limit computers are for annunciation of alarms and for administrative purposes only. This computing unit does not cause or affect a plant trip or radioactive release. (SER 91-104)

6. SPEED 91-074, New Stem and Disc Arm Materials on Aloyco Valve. The SPEED replaces Aloyco valve original stem and disc arm materials.

Summary of Safety Evaluation: The original valve stem was made of A-276-TP 316 Cond B and the disc arm was made of A-296CF8. The original valve stem to disc arm consisted of a screwed and welded connection. The original configuration is no longer available.

The new valve stem is made of A564 TP630 H1150; the new valve disc arm is made of A351-CF8; the new disc arm pin is made of A276 TP 304 Cond A. The new disc arm to stem connection is a screwed and pinned connection with the pin peened and seal welded. The disc arm and pin are both P-8 materials so a P-8 to P-8 weld procedure is acceptable. A calculation and report from the manufacturer justified the use of the materials and the new stem-to-disc arm connection. FSAR Section 6.2.2 states the motor-operated valve stems are made of A-276-TP316 Cond B or a 17-4 PH stainless steel heat treated per Westinghouse specification.

The Westinghouse specification for motor-operated valves references Specification 292082-4 for heat treatment of a 17-4 stainless steel. The difference between these specifications indicates that the Crane-Aloyco heat treatment has a valve stem that is lower in hardness RC 28 compared to Westinghouse RC 32-37. The new heat treatment is less susceptible to brittle fracture and stress corrosion cracking. The new heat treatment has tensile and yield strengths greater than A-276-TP 316 Cond B material but lower than the heat treatment specification. The lower hardness makes the new stem easier to scratch, increasing chances of packing leakage.

Based on the strength requirements being greater than A-276-TP-316 and lower susceptibility to stress corrosion cracking. The new material is acceptable. Charpy V notch testing is not required since the valves will not require operation at low temperatures. The stem requires dye penetrant testing. (SER 91-096)

7. SPEED 91-079, Replacement of Valve 2-SC-966B. Valve 2-SC 966B has excessive leakage due to a crack in the valve body. This SPEED documents the acceptability of a replacement valve. The valve has been confirmed as being an acceptable substitute by the valve manufacturer.

Summary of Safety Evaluation: This SPEED replaces applicable valve parts with identical valve components. The valve manufacturer was contacted with the existing replacement valve serial numbers and confirmed that the valves were identical, and could be interchanged. The SPEED evaluation notes that reuse of some of the existing valve components may be required to satisfy the existing valve configuration with regard to mounting and air line connections. This is considered acceptable as the valves are interchangeable. Since this is a like-for-like replacement, no unresolved safety question exists. (SER 91-102)

8. SPEED 91-080: Containment Access Door Outer O-Ring Replacement. Replacement of the outer O-ring groove on the containment access door will be 19'2" x 0.375". The door normally uses a 19'10" O-ring which is not available.

Summary of Safety Evaluation: The use of a 19'2" O-ring instead of a 19'10" O-ring in the outer groove of the containment access door seal does not present an unreviewed safety question. The SPEED demonstrates that the O-ring will provide an adequate seal. In addition, the seal will be tested by TS-10 and TS-10A prior to acceptance. The routine testing and maintenance required by TS 15.4.4 ensures that the seal remains adequate and identifies degradation prior to the seal reaching an ineffective state. (SER 91-103)

NONCONFORMANCE REPORTS (NCRs)

1. NCR N-89-116, Reactor Coolant System. NCR-N-89-116 was issued because several valve leakoffs were found to be disconnected. This situation resulted in a water spill in containment during the performance of RMP 134 during U1R16.

Summary of Safety Evaluation: FSAR Tables 6.2-11 and 6.2-12 and Sections 6.2, 9.2 and 9.3 discuss applications for valves with leakoff connections. Generally, valves that are greater than 2" nominal size, that are designated for radioactive service at an operating temperature above 212°F, that perform a modulating function or are exposed to post-accident recirculation flow, are equipped with two sets of packing with an intermediate lantern ring and leakoff connection. The fluid that leaks past the first set of packing is collected at the lantern ring and then directed to a drain collection system through the leakoff connection.

A search of the record revealed that 106 valves were originally intended to be supplied with leakoff connections directed to a drain collection system, as described in the FSAR.

At some time during the operational history of PBNP, 83 of those 106 valves were modified by having their leakoff connections disconnected. Most, but not all, of these valves had both their leakoff connections and their leakoff tubing capped. No documentation was found that may have controlled these modifications.

As there appears to be no technical problems with the 83 valves that have had their leakoff connections disabled, these valves are considered acceptable as is.

As there have been 83 undocumented modifications to these valves, the plant documentation shall be corrected to show their actual condition. Documentation changes will be required (at a minimum) to the FSAR, the P&IDs, vendor drawings, CHAMPS and the Training Handbook. The documentation changes may be controlled through the NCR process.

This evaluation is considered as an acceptable basis for future valve leakoff abandonments. The MWR process should control future valve leakoff abandonments. The MWR originator should be responsible for any future documentation updates.

Those valve leakoff tubes that have been disconnected but not capped should be capped. This may be accomplished through the MWR process. (SER 91-002)

2. NCR N-90-056, Electrical Cables. This installation replaces potentially degraded Unit 1 electrical cables. The NCR identifies a situation where several safety-related electrical cables have been subjected to a steam environment due to an overload of the condensate return unit. The cables are in trays JE06-JE07 and FV12-FV13 which were located directly above the steam vent on the El. 8' of the auxiliary building.

Summary of Safety Evaluation: During the cable replacement some of the associated cable trays will exceed the 30% fill requirement stated in the FSAR. This situation occurs when the replacement cables are installed in the cable trays while existing cables are still in place, and when the replacement cables are terminated while the old spared cables are awaiting removal. In either case, only one set of cables will be load carrying while the other set is unused. Therefore, although the cable replacement will cause some of the associated cable trays to temporarily exceed the 30% fill requirement, it will not significantly affect the ampacity or load carrying capability of the remaining cables in the cable trays. Furthermore, the cable tray supports were designed for a structural load resulting from a tray fill of 100% and therefore are not adversely affected from the cable trays temporarily exceeding the 30% fill requirement.

The new electrical cables being installed replace existing cables. The replacement cables are Class 1E and are of the same or higher quality than those being replaced. Therefore when completed, the installation will not have any affect the startup or operation of any component or system important to safety. (SER 91-035)

3. NCR N-90-056, Replacement of Certain Unit 2 Electrical Cables. The NCR identifies a situation where several safety-related electrical cables have been subjected to a steam environment due to an overload of the condensate return unit. The cables are in trays JE06 - JE07 and FV12 - FV13 which were located directly above the steam vent on the El. 8' of the primary auxiliary building.

Summary of Safety Evaluation: During cable replacement, some of the associated cable trays will exceed the 30% fill requirement stated in the FSAR. This situation occurs when the replacement cables are installed in the cable trays while existing cables are still in place, and when the replacement cables are terminated while the old spared cables are awaiting removal. In either case, only one set of cables, will be load carrying while the other set is unused. Therefore, although the cable replacement will cause some of the associated cable trays to temporarily exceed the 30% fill requirement, it will not significantly effect the ampacity or load carrying

capability of the remaining cables in the cable trays. Furthermore, the cable tray supports were designed for a structural load resulting from a tray fill of 100% and therefore are not adversely affected from the cable trays temporarily exceeding the 30% fill requirement.

The replacement cables will not be initially tied down in the cable trays. They will be tied down after the spare cables have been removed. When the replacement cables are tied down has no effect on equipment operation.

The new electrical cables replace existing cables. The replacement cables are Class 1E and are of the same or higher quality than those being replaced. Therefore when completed, the installation will not affect the startup or operation of any component or system important to safety. (SER 91-046)

V. NUMBER OF PERSONNEL AND PERSON-REM BY WORK GROUP AND JOB FUNCTION - 1991

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	68	30.890	19.030	-------	9.360	-------	0.3[illegible]0	2.130
Maintenance	44	43.270	-------	19.860	0.910	0.090	-------	22.410
Chemistry & Health Physics	36	23.280	21.510	-------	-------	-------	1.770	-------
Instrumentation & Control	16	3.980	-------	2.650	0.010	-------	-------	1.320
Technical Services	4	0.760	0.290	-------	0.150	-------	-------	0.320
Administration & Engineering, Regulatory Services	7	2.610	0.100	-------	2.510	-------	-------	-------
Utility Employees	30	26.210	0.740	10.230	3.180	-------	-------	12.060
Contractor Workers & Others	253	133.900	0.420	-------	34.610	95.630	3.240	-------
GRAND TOTALS	458	264.900	42.090	32.740	50.730	95.720	5.380	38.240

POINT BEACH NUCLEAR PLANT CALENDAR YEAR 1991		
Whole Body Exposure Range (rem)	Total Number of Individuals	Total Exposure Received (rem)
No Measurable Exposure	469	0.0
Less than 0.100	266	9.400
0.100 to 0.250	129	20.960
0.250 to 0.500	124	45.050
0.500 to 0.750	76	46.920
0.750 to 1.000	57	49.460
1.000 to 2.000	70	88.880
2.000 to 3.000	2	4.230
3.000 to 4.000	0	0.0
4.000 +	0	0.0
GRAND TOTALS	1193	264.900

591 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 18 outage, eddy current testing was performed from April 13, 1991, to April 18, 1991. An approximate 20% sample was inspected full length and an additional first support sample was done around the periphery to address loose parts concerns. As a result of findings in the "A" steam generator, an additional sample was done from the tube end hot leg to the fifth hot leg support. 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	576	584
No. 6 TSP	1416	120
No. 1 TSP	254	296
RPC	6	6
Totals	2252	1006

RPC - Rotating pancake coil inspections done to the extent necessary to bound indications.

Inspection Results: The results of these inspections showed a total of three tubes had reportable indications (1 in "B" and 2 in "A"). The cause of the indications appear to be wear at the support structure interface. As was the case during the previous inspection, several burnishing marks were noted and RPC testing was performed on a random sample of previously noted burnishing marks to monitor any growth. The signal traces from the previously noted burnishing marks were unchanged and therefore are still considered not reportable. 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%	3	3
20-29%		1
30-39%		1
60-69%	1	
MBM	46 (7)	6 (3)
MMB	2 (1)	(2)
Totals	52 (8)	11 (5)

% - Percent Through Wall Indication
MBM - Manufacturing Buff Marks
MMB - Multiple Manufacturing Buff Marks

Repaired or Plugged Tubes: The following is a list of the tubes which were mechanically plugged during the U1R18 refueling outage:

Plugged Tube in the "A" Steam Generator		
Row - Column	Indication/%	Location
21 - 63	68	-0.65" 5H

Plugged Tube in the "B" Steam Generator		
Row - Column	Indication/%	Location
30 - 60	33	AV1

5H - Fifth Support Plate Hot Leg
AV1 - #1 Anti-Vibration Bar

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 U1R18.

"A" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot Leg 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
1 - 47	MMB	1C	20.4"
1 - 47	MBM	3C	33.2"
7 - 25	MBM	5C	11.7"
8 - 56	MBM	3H	28.3"
8 - 79	MBM	3C	19.7"
11 - 73	MBM	4H	24.8"
12 - 10	MBM	4H	27.5"
12 - 26	MBM	4C	48.7"
12 - 81	MBM	4H	48.2"
12 - 90	MBM	1H	27.9"
14 - [illegible]8	MBM	1H	42.3"
14 - 63	MBM	3H	2.6"
15 - 57	MBM	5H	6.3"
15 - 62	MBM	TSH	21.8"
15 - 82	MBM	1H	10.0"
15 - 89	MBM	5C	36.1"
15 - 89	MBM	1H	18.8"
17 - 55	MMB	4H	40.2"
17 - 56	MBM	1H	30.6"
18 - 18	MBM	3C	43.5"
18 - 51	MBM	TSH	1.7"
18 - 52	MBM	TSH	2.6"
18 - 54	MBM	4H	26.0"
18 - 56	MMB	2H	22.0"

Row - Column	Indication	Location	Inch Mark
19 - 54	19	AV4	0.0"
20 - 53	MBM	3H	22.7"
23 - 17	MBM	3H	7.5"
23 - 22	MBM	3C	39.3"
23 - 45	MBM	5H	2.0"
23 - 77	MBM	2H	24.5"
24 - 8	MBM	1H	16.2"
24 - 11	MBM	3H	39.9"
24 - 12	MBM	1C	-0.5"
24 - 14	MBM	3H	45.0"
25 - 73	MBM	3H	25.2"
25 - 76	MBM	BH	8.0"
27 - 51	MBM	2C	43.0"
28 - 61	MBM	4H	5.9"
29 - 42	MBM	4H	32.8"
29 - 61	MBM	3H	2.6"
29 - 69	MBM	4H	20.8"
29 - 73	MBM	2H	25.8"
30 - 45	MBM	6H	65.3"
30 - 49	MBM	TSH	22.1"
31 - 80	18	AV3	0.0"
32 - 50	MBM	2H	11.3"
32 - 70	MBM	2H	4.7"
32 - 75	MBM	2H	49.0"
33 - 47	MBM	4H	32.0"
33 - 47	MBM	TSH	21.7"
34 - 34	MBM	2H	9.6"
35 - 52	MBM	2H	40.7"
36 - 19	16	AV4	0.0"
36 - 19	16	AV3	0.0"
36 - 32	MBM	1H	49.9"

Row - Column	Indication	Location	Inch Mark
36 - 33	MBM	3H	37.5"
36 - 35	MBM	1H	47.3"
39 - 53	MBM	6H	13.7"
40 - 37	MBM	BH	14.8"
41 - 51	MBM	5H	6.9"
42 - 42	MBM	1H	22.9"
43 - 47	MBM	4H	32.2"
44 - 40	MBM	5H	38.6"

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot Leg or Cold Leg
AV1-4 - Anti-Vibration Bar Number
BPH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

Row - Column	Indication	Location	Inch Mark
1 - 20	MBM	BC	6.7"
5 - 4	MBM	1C	51.8"
18 - 63	MBM	1H	36.7"
23 - 77	MBM	5H	5.7"
24 - 49	MMB	4C	38.0"
24 - 49	MBM	3C	10.9"
33 - 16	19	AV2	0.0"
33 - 16	18	AV3	0.0"
33 - 22	MBM	AV2	16.7"
33 - 71	13	AV1	0.0"
34 - 75	17	AV2	0.0"
35 - 51	13	AV1	0.0"
35 - 51	20	AV2	0.0"
36 - 37	MBM	5H	-0.6"
39 - 38	MBM	4C	19.6"
39 - 63	MBM	BH	12.5"
41 - 44	MMB	BC	18.1"

Row - Column	Indication	Location	Inch Mark
41 - 46	MBM	BH	1.6"

UNIT 2

Inspection Results: During the Unit 2 Refueling 17 outage, eddy current was performed October 4, 1991, to October 11, 1991. An approximate 20% sample (including sleeved and unsleeved tubes) was full length inspected, the unsleeved hot leg tubes were inspected to the first hot leg support plate, the entire cold leg of the "B" steam generator not covered by the full length program was tested to the first cold leg support plate, and a sample of previously defective and their surrounding tubes in the "A" steam generator cold leg were tested to the first cold leg support plate. The extended tested in each steam generator is as follows:

Eddy Current Inspection Plan		
Extent of Inspection	Number of Tubes	
	"A" SG	"B" SG
Full Length	303	331
No. 1 TSP	1147	3062
CL Sleeves	24	155
HL Sleeves	300	281
From Tube End CL to Sleeve Top HL	302	282
RPC	35	26
Totals	2376	4111

RPC - Rotating pancake coil inspections done to the extent necessary to bound distorted indications.

Inspection Results: The results of these inspections revealed 65 tubes in the "A" steam generator with reportable indications and 107 in the "B" steam generator. The cause of the majority of the indications in the hot legs of both steam generators is suspected to be crevice corrosion. The support plate indications are assumed to be IGA/SCC originating from deposits between the tube and tube support plate. The cold leg indications all appear to be a result of wastage. Comparisons have been made from the previous inspection data and the average growth rate is maintaining itself at the 5% per year rate as was previously noted. Also continuing is the deterioration of the hot leg crevices from no detectable indication to pluggable indications. 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 (1)	7 (2)
DRI	2	
0-19%	5 (6)	4 (31)
20-29%	4 (4)	4 (42)
30-39%	1 (4)	4 (25)
40-49%		2
50-59%	1	(1)
60-69%	1	2
70-79%	1	
80-89%	1	
90-99%	1	
S/MAI	30	20
SQR	12	4
Totals	65 (15)	47 (101)

% - Percent Through Wall Indication
DI - Distorted Indication
DRI - Distorted Roll Indication
S/MAI Single/Multiple Axial Indication
SQR - Squirrel Indication

Repaired or Plugged Tubes: The following lists the tubes which were mechanically plugged during the Unit 2 Refueling 17 outage:

NOTE: All inch marks are above the referenced location unless otherwise specified.

Plugged Tubes in the "A" Steam Generator		
Row - Column	Indication/%	Location
2 - 4	SQR	3.6" TEH
21 - 9	SQR	4.9" TEH
19 - 10	SQR	3.8" TEH
3 - 12	MAI	6.7" TEH
32 - 21	SAI	6.6" TEH
35 - 22	SQR	2.7" TEH
37 - 23	MAI	3.9" TEH
35 - 30	MAI	4.6" TEH
40 - 31	SQR	3.1" TEH
41 - 31	MAI	5.2" TEH
39 - 33	SAI	6.2" TEH
35 - 34	MAI	5.1" TEH
37 - 34	SAI	5.1" TEH
40 - 35	MAI	5.1" TEH
41 - 36	SAI	7.8" TEH
40 - 37	SAI	4.6" TEH
39 - 37	SAI	4.7" TEH
35 - 38	SAI	4.9" TEH
35 - 39	MAI	3.3" TEH
34 - 40	MAI	4.3" TEH
41 - 44	SAI	4.6" TEH
42 - 45	SAI	5.5" TEH
42 - 46	MAI	7.4" TEH
40 - 47	SAI	6.1" TEH
43 - 51	MAI	6.6" TEH
34 - 59	SQR	5.1" TEH
35 - 59	SQR	8.2" TEH
42 - 64	65%	1H

Plugged Tubes in the "A" Steam Generator		
Row - Column	Indication/%	Location
35 - 70	55%	3.5" TEH
33 - 72	MAI	5.6" TEH
4 - 72	SQR	9.9" TEH
33 - 74	SQR	3.0" TEH
32 - 76	SQR	4.0" TEH
32 - 78	SAI	5.6" TEH
14 - 79	MAI	13.3" TEH
20 - 79	SAI	12.7" TEH
15 - 79	SAI	9.9" TEH
4 - 80	SAI	4.3" TEH
25 - 81	SQR	4.4" TEH
24 - 81	DRI	2.3" TEH
1 - 82	97%	2.9" TEH
14 - 83	SQR	3.3" TEH
19 - 84	SAI	4.9" TEH
4 - 85	SAI	3.4" TEH
6 - 85	SAI	7.1" TEH
14 - 87	82%	4.0" TEH
9 - 87	MAI	7.4" TEH
12 - 90	72%	4.6" TEH
11 - 90	DRI	2.2" TEH

Plugged Tubes in the "B" Steam Generator		
Row - Column	Indication/%	Location
7 - 5	SAI	1.8" TEH
1 - 8	SQR	3.7" TEH
11 - 8	SQR	10.7" TEH
13 - 8	44%	6.4" TEH
20 - 9	SQR	6.4" TEH
6 - 9	SQR	2.8" TEH
20 - 13	SAI	6.8" TEH

Plugged Tubes in the "B" Steam Generator		
Row - Column	Indication/%	Location
17 - 16	MAI	4.9" TEH
3 - 38	47%	2H
3 - 57	53%	0.4" TSC
37 - 58	61%	10.2" TEH
39 - 59	SAI	5.8" TEH
37 - 59	61%	6.5" TEH
36 - 61	MAI	5.8" TEH
36 - 62	MAI	8.6" TEH
39 - 62	SAI	7.8" TEH
33 - 68	SAI	3.6" TEH
36 - 69	SAI	5.2" TEH
2 - 71	MAI	4.4" TEH
24 - 74	SAI	4.5" TEH
24 - 74	SAI	5.2" TEH
1 - 76	SAI	4.0" TEH
7 - 77	SAI	4.4" TEH
1 - 81	SAI	3.5" TEH
17 - 82	MAI	9.6" TEH
17 - 83	SAI	5.3" TEH
14 - 84	MAI	5.3" TEH
21 - 85	MAI	6.2" TEH
10 - 85	SAI	8.8" TEH
16 - 88	SAI	5.9" TEH

1H/1C - Support Plate Number Hot or Cold Leg
TEH - Tube End Hot Leg
TSC - Tubesheet Cold Leg

Tubes with Indications - Not Plugged: The following is a list of tubes with indications not repaired during Unit 2 Refueling 17 outage as a result of eddy current indications:

"A" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot Leg or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Tube End 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 - 91	DI	1C	0.0"
2 - 10	DI	TEH	4.0"
3 - 77	DI	TSH	0.3"
6 - 25	25	TSC	0.6"
6 - 33	13	TSC	0.4"
7 - 1	38	1C	0.0"
7 - 33	15	TSC	0.5"
7 - 34	21	TSC	0.6"
7 - 37	25	TSC	0.5"
7 - 78	15	TSH	0.6"
8 - 34	35	TSC	0.7"
8 - 36	33	TSC	0.5"
9 - 37	33	TSC	0.4"
10 - 25	11	TSC	0.6"
14 - 39	13	TSC	0.7"
18 - 5	23	1H	0.0"
18 - 6	38	1H	0.0"
18 - 60	15	TSC	11.3"
18 - 63	12	TSC	7.3"
20 - 37	26	TSC	0.6"
22 - 7	27	2H	0.0"
23 - 76	DI	TEH	11.7"
23 - 76	DI	TEH	8.7"
31 - 16	DI	TEH	8.8"

Row - Column	Indication	Location	Inch Mark
34 - 75	DI	TEH	17.0"
36 - 59	20	TSH	3.6"
38 - 33	DI	TSH	0.3"
40 - 26	20	TSH	8.2"
40 - 26	19	TSH	5.6"
40 - 58	[illegible]	TSH	9.4"
42 - 56	15	TSH	10.1"
43 - 44	12	1H	0.0"

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot Leg or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Tube End Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

Row - Column	Indication/%	Location	Inch Mark
1 - 2	30	1C	0.0"
1 - 21	16	1C	0.0"
1 - 22	14	1C	0.0"
1 - 27	DI	TEH	6.3"
1 - 52	29	TSH	0.8"
1 - 62	DI	TEH	3.4"
2 - 1	13	TSC	8.9"
2 - 48	27	TSC	0.6"
3 - 1	22	1C	0.0"
3 - 55	10	TSC	0.7"
5 - 2	24	1C	0.0"
6 - 16	22	TSH	0.5"
7 - 1	16	TSH	10.1"
8 - 91	17	TSH	0.7"
9 - 81	32	1H	20.9"
11 - 29	10	TSC	2.6"
11 - 29	14	TSC	0.8"

Row - Column	Indication/%	Location	Inch Mark
12 - 2	30	1C	0.0"
12 - 72	30	TSC	0.7"
13 - 4	32	1C	0.0"
15 - 33	19	TSC	0.8"
17 - 22	18	1C	0.0"
17 - 36	19	TSC	[illegible]
18 - 12	16	TSH	[illegible]
19 - 37	26	TSC	0.8"
20 - 58	32	TSC	0.7"
20 - 68	12	TSC	0.6"
21 - 7	26	1C	0.0"
21 - 7	31	1C	0.0"
22 - 24	24	1C	0.0"
22 - 28	34	1C	0.0"
22 - 77	DI	TEH	3.5"
22 - 86	33	1C	0.0"
22 - 86	33	1C	0.0"
23 - 29	31	1C	0.0"
23 - 69	39	1C	0.0"
24 - 21	37	TSH	33.6"
24 - 44	21	TSC	0.6"
26 - 13	28	1C	0.0"
26 - 24	21	1C	0.0"
26 - 51	38	1C	0.0"
26 - 75	17	TSC	25.2"
27 - 29	26	1C	0.0"
27 - 51	39	1C	0.0"
27 - 69	33	1C	0.0"
28 - 16	26	TSH	47.0"
28 - 39	27	1C	0.0"
28 - 51	23	1C	0.0"

Row - Column	Indication/%	Location	Inch Mark
29 - 16	28	1C	0.0"
29 - 19	26	1C	0.0"
29 - 26	30	1C	0.0"
29 - 28	19	1C	0.0"
29 - 30	DI	1C	0.0"
30 - 36	36	1C	0.0"
30 - 48	28	1C	0.0"
30 - 49	16	1C	0.0"
30 - 50	19	1C	0.0"
31 - 28	24	1C	0.0"
31 - 66	39	1C	0.0"
31 - 67	19	1C	0.0"
32 - 20	22	1C	0.0"
32 - 21	18	1C	0.0"
32 - 60	39	1C	0.0"
32 - 63	20	1C	0.0"
32 - 63	DI	TEH	11.7"
32 - 70	29	1C	0.0"
33 - 19	16	1C	0.0"
33 - 37	DI	TSH	0.2"
33 - 46	23	1C	0.0"
33 - 48	29	1C	0.0"
33 - 48	24	TSH	0.6"
33 - 58	21	1C	0.0"
33 - 60	28	1C	0.0"
33 - 63	DI	TEH	7.5"
33 - 71	27	1C	0.0"
33 - 72	25	1C	0.0"
33 - 73	28	1C	0.0"
33 - 74	DI	1C	0.0"
34 - 23	19	1C	0.0"

Row - Column	Indication/%	Location	Inch Mark
34 - 39	DI	TSH	0.0"
34 - 39	34	1C	0.0"
34 - 67	37	1C	0.0"
35 - 70	34	1C	0.0"
36 - 19	30	1C	0.0"
36 - 21	18	1C	0.0"
36 - 22	22	1C	0.0"
36 - 25	26	1C	0.0"
36 - 31	19	1C	0.0"
36 - 48	18	TSH	41.5"
36 - 48	18	TSH	35.6"
36 - 63	20	1C	0.0"
36 - 65	16	1C	0.0"
36 - 66	26	1C	0.0"
37 - 21	[illegible]	1C	0.0"
37 - 23	26	1C	0.0"
37 - 25	22	1C	0.0"
37 - 28	24	1C	0.0"
37 - 61	31	1C	0.0"
37 - 62	31	1C	0.0"
37 - 63	21	1C	0.0"
37 - 67	21	1C	0.0"
37 - 68	17	1C	0.0"
37 - 73	24	1C	0.0"
38 - 28	18	TSC	31.6"
38 - 49	14	1C	0.0"
38 - 52	31	1C	0.0"
38 - 54	28	1C	0.0"
38 - 61	32	AV4	0.0"
39 - 25	27	1C	0.0"
39 - 34	30	1C	0.0"

Row - Column	Indication/%	Location	Inch Mark
39 - 61	20	1C	0.0"
39 - 63	14	1C	0.0"
39 - 64	21	1C	0.0"
39 - 65	17	1C	0.0"
40 - 57	14	1C	0.0"
41 - 46	18	1C	0.0"
41 - 47	18	1C	0.0"
41 - 48	18	1C	0.0"
45 - 47	35	2H	0.0"

VII REACTOR COOLANT SYSTEM RELIEF VALVE CHALLENGES

Overpressure Protectio: 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 1991.

Overpressure Protection During Normal Pressure and Temperature Operation

There were no challenges to Unit 1 or Unit 2 power-operated relief valves during low temperature and low pressure operation in 1991.

VIII. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications during operation of Unit 1 and Unit 2 in 1991 where reactor coolant activity exceeded that allowed by Technical Specifications.