WISCONSIN ELECTRIC

POWER COMPANY

ANNUAL RESULTS AND

DATA REPORT

1993

POINT BEACH NUCLEAR PLANT

UNIT NOS. 1 AND 2

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 1993 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. 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, 1993, through December 31, 1993, included a 40-day refueling/maintenance outage. Major work items included inspection and eddy current testing of the steam generators; mechanical and electrical upgrades of the main steam isolation valves (MSIVs); installation of rod drop testing equipment; amptector upgrades of 480 V bus feeder and MG set breakers; P30B circulating water pump inspection; and as-built electrical walkdowns of the reactor protection and safeguards systems. A steam generator feedwater pump shaft-driven lube oil pump was repaired during the year. Molded case circuit breakers were replaced as part of the upgrade project.

Unit 1 operated at an average capacity factor of 89.5% (MDC net) and an electrical/thermal efficiency of 32.4%. The unit and reactor availability were 89.1% and 89.4%, respectively. Unit 1 generated its 75 billionth kilowatt hour on January 11, 1993; its 76 billionth kilowatt hour on May 14, 1993; its 77 billionth kilowatt hour on August 3, 1993; and its 78 billionth kilowatt hour on October 23, 1993.

UNIT 2

Highlights for the period January 1, 1993, through December 31, 1993, included a 35-day refueling/maintenance outage. Major work items included inspection and eddy current testing of the steam generators; inspection of the high pressure turbine; replacement of breakers in safeguards MCC 2B32 and 2B42; as-built walkdowns of electrical distribution buses; reactor coolant pump seal inspection and replacement; and replacement of 21 480 V safety-related molded case circuit breakers. Power was reduced for replacement of a failed pressure transmitter that provides an input signal to the feedwater turbine trip circuit; condenser waterbox inspection for tube leakage because of increased sodium levels in the steam generators; and, inspection and repair of a main condensate pump casing.

Unit 2 operated at an average capacity factor of 90.5% (MDC net) and an electric/thermal efficiency of 32.8%. The unit and reactor availability were 90% and 90.5%, respectively. Unit 2 generated its 76 billionth kilowatt hour on March 22, 1993; its 77 billionth kilowatt hour on June 13, 1993; its 78 billionth kilowatt hour on September 2, 1993; and its 79 billionth kilowatt hour on December 30, 1993.

III. Amendments to Facility Operating Licenses

During 1993, there were seven amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1 and seven 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 137 to DPR-24, Amendment 141 to DPR-27, March 26, 1993: The amendments modified the safeguards bus degraded voltage protection relay setpoint to remove nonconservatisms in the setpoints.

Amendment 138 to DPR-24, Amendment 142 to DPR-27, April 16, 1993: The amendments removed requirements to test the operable train safeguards system equipment when the other train is discovered inoperable or is otherwise removed from service.

Amendment 139 to DPR-24, Amendment 143 to DPR-27, April 20, 1993: The amendments modified the limiting condition for operation for safety injection accumulators to allow one hour to return an accumulator to an operable status when inoperable for any reason.

Amendment 140 to DPR-24, Amendment 144 to DPR-27, July 28, 1993: The amendments increased the interval for surveillance testing of reactor protection system and engineered safeguards system analog instrumentation from monthly to quarterly and removed requirements to log selected other instrumentation when the unit is in a cold shutdown condition.

Amendment 141 to DPR-24, Amendment 145 to DPR-27, September 7, 1993: The amendments modified test requirements for safety injection system tests to allow the tests to be conducted with the safety injection and residual heat removal pump motor breakers in test or racked in and operable.

Amendment 142 to DPR-24, Amendment 146 to DPR-27, October 27, 1993: The amendments lowered the Unit 2 reactor coolant system flow limits to account for higher steam generator tube plugging levels.

Amendment 143 to DPR-24, Amendment 147 to DPR-27, December 26, 1993: The amendments added and clarified the limiting condition for operation and surveillance requirements for the main steam stop valves and non-return check valves.

IV. 10 CFR 50.59

PROCEDURE CHANGES

1. ECA-0.0, (Major), Loss of All AC Power, Revision 11. (Permanent)

   Summary of Safety Evaluation: Format changes were made in accordance with the EOP writer's guide. This was the first EOP to use the unit specific format. Action steps were revised to provide guidance to reduce the amount of time reactor coolant pump (RCP) seal leakage remains unisolated following a loss of AC power. This was accomplished by incorporating only those actions which can be accomplished from the control room to initially restore 480 Vac safeguards power. Local actions are accomplished through a series of appendices initiated after RCP seal isolation. Guidance was provided for modifications to the dc distribution system and use of alternate power sources for selected shutdown loads such as service water pumps and charging pumps from buses B08 and B09.
   (SER 88-091-10)

2. EOP-1.3, (Major), Transfer to Containment Sump Recirculation, Revision 10. (Temporary)

Summary of Safety Evaluation: A note was added prior to changing valve lineup to transfer to containment sump recirculation to state that CSPs should not be implemented until step is complete. A note was added prior to Step 23 and Step 15a RNO is changed to implement CSPs as required when completing or transitioning from the valve lineup. Containment sump recirculation is considered maintenance of core cooling and should be performed prior to this transition. (SER 88-089-12)

3. EOP-1.4, (Major), Transfer to Containment Sump Recirculation, One Train Inoperable, Revision 6. (Temporary)

Summary of Safety Evaluation: A note was added prior to changing valve lineup to transfer to containment sump recirculation to state that CSPs should not be implemented until the step is complete. Step 10 was added to verify adequate cooling as in EOP-1.3. A note prior to Step 10 and a change to Step 9f RNO ensures CSPs are implemented as required when completing or transitioning from the valve lineup. Containment sump recirculation is considered maintenance of core cooling and should be performed prior to this transition. (SER 89-100-12)

4. ECA-3.1, Unit 1, (Major), Steam Generator Tube Rupture (SGTR) With Loss of Reactor Coolant-Saturated Recovery Desired, Revision 12. (Permanent)

Summary of Safety Evaluation: The changes to ECA-3.2 allow the last SI pump to be secured based on having a residual heat removal (RHR) pump running, subcooling and pressurizer level. The condition of RCS >[270°F] 310°F was deleted since this procedure deals with saturated recovery. The [270°F] 310°F was calculated for subcooled recovery and are is used in ECA-3.1. Without this change, the SI pump is not secured until CSP-P.1 entry is made which is likely to occur after the steam generator relief valve has lifted due to SI discharge pressure. Core cooling is still provided by the RHR pump, but the shutoff head of the RHR pump will not lift a steam generator relief valve. (SER 89-034-11)

5. HP 3.2.5, (Minor), Radioactive Material Posting Requirements. (Permanent)

Summary of Safety Evaluation: Materials stored in Warehouse 2 mainly consist of tools and spare parts used once or periodically with long periods of non-use. These materials contain low-level fixed and loose contamination. The area is not to be used for radioactive waste storage. The hazard posed by the fixed portion of the contamination was evaluated in SER 92-011 and addresses the hazard posed by the loose contamination.

The unreviewed safety question evaluation (SER 92-011) applies to the storage of materials with loose contamination in Warehouse 2 and included the following additions: The facility design must be justified on the basis of 10 CFR 20 and 10 CFR 100 requirements. A hazards analysis was performed. It was determined that the destruction of a package in the storage area is extremely unlikely to produce a radiological consequence that would exceed a small fraction of 10 CFR 20 or of 10 CFR 100 limits.

An additional engineering control is imposed because the byproduct materials are in the form of loose contamination and are stored in an area without effluent controls. The potential for release exists which could lead to the exposure of plant personnel and members of the general public that is not ALARA. All materials in the storage area shall be stored in strong, tight packaging. Strong, tight packaging is defined as packaging that will not leak any of the radioactive material during conditions incident to the storage.

The equipment to be stored in the area shall be stored in a locked, secured area that guards against unauthorized removal during the term of the storage. Additionally, the materials shall be stored in packaging that does not support combustion. (SER 93-005)

6. 1ICP-02.005, Unit 1, (Major), Engineered Safety Features System Logic Monthly Surveillance Test, Revision 0. (New Procedure)

1ICP-02.005A-1, Unit 1, (Minor), Engineered Safety Features System Logic Train A Monthly Surveillance Test, Revision 0. (New Procedure)

1ICP-02.005B-1, Unit 1, (Minor), Engineered Safety Features System Logic Train B Monthly Surveillance Test, Revision 0. (New Procedure)

2ICP-02.005, Unit 2, (Major), Engineered Safety Features System Logic Monthly Surveillance Test, Revision 0. (New Procedure)

2ICP-02.005A-1, Unit 2, (Minor), Engineered Safety Features System Logic Train A Monthly Surveillance Test, Revision 0. (New Procedure)

2ICP-02.005B-1, Unit 2, (Minor), Engineered Safety Features System Logic Train B Monthly Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: The revisions included a technical review of Technical Specifications, FSAR, CHAMPS, and electrical and logic drawings. Steps were added to list affected annunciators and status lights with operator notification before each step is entered that affects an annunciator or status light. Conditions necessary to verify final restoration to service by the DSS were included.

The tests required opposite train components to be operable. Combined with the 1/2 nature of the system, relaxation of single failure criterion during bypass is permitted in accordance with FSAR Page 7.2.13. Test methodology is not changed from that of previous tests. The tests are not described in the FSAR; however, a proposed change to Technical Specifications states that for I&C relay logic testing "Automatic isolation functions of tested instrumentation and controls shall not be declared incapable of performing their related safety functions providing that: 1. The entire test be completed in one shift. 2. Instrumentation and controls are restored to service within one shift. 3. No instrumentation and controls are discovered to be inoperable."

Operations Special Order, PBNP 92-02, permits safeguards testing while at power for a single shift. (SER 93-011)

7. 1ICP-02.005, Unit 1, (Major), Engineered Safety Features System Logic Monthly Surveillance Test, Revision 1. (Permanent)

2ICP-02.005, Unit 2, (Major), Engineered Safety Features System Logic Monthly Surveillance Test, Revision 1. (Permanent)

1ICP-02.025, Unit 1, (Major), Engineered Safety Features System Logic Five Year Surveillance Test, Revision 1. (Permanent)

2ICP-02.025, Unit 2, (Major), Engineered Safety Features System Logic Five Year Surveillance Test, Revision 1. (Permanent)

Summary of Safety Evaluation: The procedures direct engineered safety features (ESF) surveillance testing. The revision removes specific prerequisite component conditions thought to be necessary because of performance of this surveillance. The first of these was

"All equipment in engineered safety features Train B shall be operable." The second was "If one or more service water pumps in the train opposite that to be tested is out of service, at least the same number of pumps in the train to be tested shall be running." These component conditions are replaced by a single statement, "No Train A (Train B for the other procedure) emergency safety features system LCOs are in effect." Other specific conditions, arising because of lessons learned, commitments, or good practice still remain.

During testing, the service water cooling load isolation logic channel is placed in a condition simulating no service water pumps in operation. If this occurred during testing, 30 seconds after an SI signal, service water loads would be isolated.

A high steam generator level logic channel that isolates feedwater to one SG loop is disabled during testing. The untested train initiates feedwater isolation in that loop, and the opposite loop logic channels are operable.

These are routine surveillance activities. The phrase "No Train A engineered safety features system LCOs are in affect," reflects routine plant conditions. At no time does performance of this surveillance place the plant outside of analyzed conditions. With the exception of containment spray, sufficient initiating signal diversity exists to assure that a required automatic function in the tested train occurs during testing. For containment spray, there is a redundant train, and accident fan coolers provide further cooling. Further, the occurrence of a design basis LOCA or SLB coincident with test performance is to a high credibility scenario. (SER 93-011-01)

8. 1ICP-02.008, Unit 1, (Major), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

1ICP-02.008-1, Unit 1, (Minor), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

1ICP-02.008-2, Unit 1, (Minor), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

2ICP-02.008, Unit 2, (Major), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

2ICP-02.008-1, Unit 2, (Minor), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

2ICP-02.008-2, Unit 2, (Minor), Nuclear Instrumentation Power Range Axial Offset Calibration, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specifications, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walkdown of accessible components, commitments and CHAMPS.

The tests are required by TS Table 15.4.1-1 and FSAR Section 7.4.3. Test methodology does not vary from that specified in the FSAR nor with that of previous performances of these activities. (SER 93-043)

9. 1ICP-02.009, Unit 1, (Major), Nuclear Instrumentation Immediate Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

1ICP-02.009-1, Unit 1, (Minor), Nuclear Instrumentation Immediate Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.009, Unit 2, (Major), Nuclear Instrumentation Immediate Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.009-1, Unit 2, (Minor), Nuclear Instrumentation Immediate Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specifications, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walkdown of accessible components, commitments and CHAMPS.

These tests are required by TS Table 15.4.1-1 and FSAR Section 7.4.3. This activity has no effect on accident probability. Test methodology does not vary from that specified in the FSAR nor with that of previous performances of these activities. (SER 93-044)

10. 1ICP-02.010, Unit 1, (Major), Nuclear Instrumentation Source Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

1ICP-02.010-1, Unit 1, (Minor), Nuclear Instrumentation Source Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.010, Unit 2, (Major), Nuclear Instrumentation Source Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.010-1, Unit 2, (Minor), Nuclear Instrumentation Source Range Pre-Startup Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specifications, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walkdown of accessible components, commitments and CHAMPS.

These tests are required by TS Table 15.4.1-1 and FSAR Section 7.4.3. The activities have no effect on accident probability. At least one source range channel is in service at all times. Test methodology does not vary from that of previous performances of these activities. (SER 93-046)

11. 1ICP-02.014, Unit 1, (Major), Nuclear Instrumentation Power Range Pre-Reactor Startup Surveillance Test, Revision 0. (New Procedure)

1ICP-02.014-1, Unit 1, (Minor), Nuclear Instrumentation Power Range Pre-Reactor Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.014, Unit 2, (Major), Nuclear Instrumentation Power Range Pre-Reactor Startup Surveillance Test, Revision 0. (New Procedure)

2ICP-02.014-1, Unit 2, (Minor), Nuclear Instrumentation Power Range Pre-Reactor Startup Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specification, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walkdown of accessible components, commitments and CHAMPS.

The tests are required by TS Table 15.4.1-1 and FSAR Section 7.4.3. These activities have no effect on accident probability. The tests are performed while the unit is shutdown and on one power range channel at a time. Each channel under test is placed back in service before proceeding to test the next channel. Test methodology does not vary from that of previous performances of these activities. (SER 93-048)

12. 1ICP-02.019, (Major), Engineered Safety Features System Logic Shutdown Surveillance Test, Revision 0. (New Procedure)

1ICP-02.019-1, (Minor), Engineered Safety Features System Logic Trains A and B Shutdown Surveillance Test, Revision 0. (New Procedure)

2ICP-02.019, (Major), Engineered Safety Features System Logic Shutdown Surveillance Test, Revision 0. (New Procedure)

2ICP-02.019-1, (Minor), Engineered Safety Features System Logic Trains A and B Shutdown Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specifications, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walk-down of accessible components, commitments and CHAMPS.

The tests are required by Technical Specifications Table 15.4.1-1, Items 7, 11, 18, 24, and 25, and have no effect on accident probability. The tests are performed while the unit is in cold shutdown. (SER 93-082)

13. 1ICP-02.024, Unit 1, (Major), Reactor Protection System Logic Five Year Surveillance Test, Revision 0. (New Procedure)

1ICP-02.024A-1, Unit 1, (Minor), Reactor Protection System Logic Train A Five Year Surveillance Test, Revision 0. (New Procedure)

1ICP-02.024B-1, Unit 1, (Minor), Reactor Protection System Logic Train B Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.024, Unit 2, (Major), Reactor Protection System Logic Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.024A-1, Unit 2, (Minor), Reactor Protection System Logic Train A Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.024B-1, Unit 2, (Minor), Reactor Protection System Logic Train B Five Year Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review of Technical Specifications, FSAR, CHAMPS, and electrical and logic drawings was performed.

The appropriate reactor trip bypass breaker is tested operable in the test position. The reactor trip bypass breaker is then fully racked in, bypassing the reactor trip breaker. The logic channel to be tested is selected at the logic test panel. Logic channel testing is then performed using the first trip matrix tested to trip the reactor trip breaker and verifying the use of the test light on the logic test panel. The remaining logic channel trips are tested using the logic test panel test light. All annunciators, status lights, and computer points are checked for reliability of indication. Only one reactor protection system train is tested at a time and the other train is declared operable before testing begins. The train not under test has the required minimum operable channels set forth in TS Table 15.3..5-2 and can perform the intended reactor trip function. The testing methodology analyzed in FSAR Section 7.2 is not changed. (SER 93-010)

14. 1ICP-02.025, Unit 1, (Major), Engineered Safety Features System Logic Five Year Surveillance Test, Revision 0. (New Procedure)

1ICP-02.025A-1, Unit 1, (Minor), Engineered Safety Features System Logic Train A Five Year Surveillance Test, Revision 0. (New Procedure)

1ICP-02.025B-1, Unit 1, (Minor), Engineered Safety Features System Logic Train B Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.025, Unit 2, (Major), Engineered Safety Features System Logic Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.025A-1, Unit 2, (Minor), Engineered Safety Features System Logic Train A Five Year Surveillance Test, Revision 0. (New Procedure)

2ICP-02.025B-1, Unit 2, (Minor), Engineered Safety Features System Logic Train B Five Year Surveillance Test, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review of Technical Specifications, FSAR, CHAMPS, and electrical and logic drawings was performed.

The tests are performed by momentarily disabling a single logic channel to a single safety feature, performing matrix and single point testing, checking all annunciators, status lights, and computer points for reliability of indication, restoring the channel to service, and confirming test switch operation by means of a continuity check. The tests require all opposite train components be operable. Combined with the 1/2 nature of this system, the relaxation of single failure criterion during bypass is permitted. Test methodology is not changed from that of previous tests.

The tests are not described in the FSAR; however, a proposed change to Technical Specifications states that for I&C relay logic testing, "Automatic isolation functions of tested instrumentation and controls shall not be declared incapable of performing their related safety functions providing that: 1. The entire test be completed in one shift. 2. Instrumentation and controls are restored to service within one shift. 3. No instrumentation and controls are discovered to be inoperable."

Operations Special Order PBNP 92-02 permits safeguards testing while at power for a single shift. (SER 93-012)

15. 1ICP-10.001, (Major), Engineered Safety Features System and AMSAC System Bypass, Revision 0. (New Procedure)

1ICP-10.001-1, (Minor), Engineered Safety Features System and AMSAC System Bypass, Revision 0. (New Procedure)

2ICP-10.001, (Major), Engineered Safety Features System and AMSAC System Bypass, Revision 0. (New Procedure

2ICP-10.001-1, (Minor), Engineered Safety Features System and AMSAC System Bypass, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures include a complete technical review, including cross-review with Technical Specifications, FSAR, associated drawings, earlier versions of the existing procedure, existing procedure history file, vendor information, a walk-down of accessible components, commitments and CHAMPS.

The procedures bypass AMSAC and engineered safety features system automatic safety injection circuits by placing test switches to the test position. Automatic containment isolation and containment ventilation isolation are bypassed as a result of safety injection (SI) signals being bypassed. The procedures are not required by Technical Specifications, but are performed to allow testing and calibrations during cold shutdown without the possibility of creating a SI signal. Switch continuity testing is performed to insure switch operability. The activities have no effect on accident probability. (SER 93-085)

16. 2ICP-11.456, (Major), Functional Test of Nuclear Instrumentation Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.456-1, (Minor), Functional Test of Nuclear Instrumentation Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures include a technical review including Technical Specifications, FSAR, associated drawings and CHAMPS. The tests are not a complete functional test of nuclear instrumentation (NI) but only of circuits connected through 2C158. All three ranges of NIs are included. Steps are included for setup prior to testing each NI range. Appropriate precautions have been made not to violate Technical Specifications during shutdown. Source range high flux at shutdown containment horn will be disabled during source range testing. Prerequisites and steps are included to ensure no refueling operations and at least one source range is operable before testing begins on a source range channel. Steps were included to switch audio count rate to the operating source range before testing begins. During power range testing automatic rod stops are disabled to facilitate circuit testing. No rod movement is a prerequisite of performance.

Steps were added listing conditions necessary to verify equipment has been returned to service prior to obtaining the duty shift superintendent's signature. These test procedures do not result in a reduction in the margin of safety since adequate safety and control functions are maintained during testing. (SER 93-064)

17. 2ICP-11.457, (Major), Functional Test of Chemical and Volume Control Instrumentation Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.457-1, (Minor), Functional Test of Chemical and Volume Control Instrumentation Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures include a complete technical review including Technical Specifications, FSAR, all associated drawings and CHAMPS. The tests are not a complete functional test of chemical and volume control (CVCS) instrumentation but only of circuits connected through 2C158. The tests are performed during cold shutdown conditions. A prerequisite of the volume control tank being bypassed was included to prevent any abnormal flowpaths that could be generated if there was purification flow. The procedures do not alter the boric acid injection flowpath required by TS 15.3.2.A. Steps were added listing conditions necessary to verify the equipment was returned to service prior to obtaining the duty shift superintendent (DSS) signature. (SER 93-065)

18. 2ICP-11.458, (Major), Functional Test of T_{AVG} Loops For Condenser Steam Dump Control Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.458-1, (Minor), Functional Test of T_{AVG} Loops For Condenser Steam Dump Control Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures include a technical review including Technical Specifications, FSAR, associated drawings and CHAMPS. The tests are not a complete functional test of T_{AVG} instrumentation but only of condenser steam dump control circuits connected through 2C158. Independent verification is used when opening/closing sliders and during the removal and installation of fuses. The tests are performed during cold shutdown conditions. A prerequisite condition of no turbine stop valve stroking is a requirement for test setup only. Steps were added listing conditions necessary to verify the equipment was returned to service prior to obtaining the DSS signature. The performance of these procedures do not constitute an unreviewed safety question. (SER 93-066)

19. 2ICP-11.460, (Major), Functional Test of Residual Heat Removal (RHR) Valve Interlocks, Low Power Auto Rod Withdrawal Stop, and Component Cooling (CC) Water System Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.460-1, (Minor), Functional Test of Residual Heat Removal (RHR) Valve Interlocks, Low Power Auto Rod Withdrawal Stop, and Component Cooling (CC) Water System Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review was performed including Technical Specifications, FSAR, associated drawings and CHAMPS.

The tests are not a complete functional test of component cooling (CC) pumps standby start feature on low header pressure but only circuits connected through 2C158. The functions of 2PC-639-X1 and 2PC-639-X2, automatic start of CC pumps on low header pressure are tested by manually stopping 2P11B, CC pump causing low header pressure and observing 2P11A, CC pump start and annunciation occurring. Relay 2PC-639-X2 is tested by observing the plunger pull in and drop out. The test is performed during cold shutdown. Having 2P11B in service is a prerequisite. 2P11A will start as a result of this test. Flow in the CC system is checked during test.

The procedures do not result in a reduction of the margin of safety. The procedures are performed during shutdown conditions. Both CC pumps are maintained in service providing CC flow to the chemical and volume control system (CVCS) non-regenerative heat exchangers for RHR. The margin of safety is not reduced since adequate safety and control functions are maintained during and ensured after testing. (SER 93-067)

20. 2ICP-11.461, (Major), Functional Test of Pressurizer Low Level and High Pressure Loops Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.461-1, (Minor), Functional Test of Pressurizer Low Level and High Pressure Loops Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review of the Technical Specifications, FSAR, associated drawings and CHAMPS was performed. The tests are not a complete functional test of pressurizer heaters control instrumentation but only that portion of the circuits connected within 2C158. The tests are performed when the reactor is in cold shutdown and depressurized. Normal letdown isolation valve 2RC-427, is shut during test but is upstream of 2CV-371A which is closed when letdown flow for purification is provided by the residual heat removal (RHR) system during shutdown conditions. The tests do not cause or create any drainage flowpaths of reactor coolant. Steps for testing automatic circuits for pressurizer heaters are included. A prerequisite for pressurizer heaters to be racked out and red tagged is added.

No new equipment was installed or existing equipment modified. This is verified through functional testing of all affected circuitry at the completion of testing. Verification of instrument signal plug removals and opening of sliders are performed before testing begins and again when signal plugs are installed and sliders are closed ensuring the tested equipment is returned to service. Steps were added to verify that testing of all circuits was completed and lists conditions necessary to verify the equipment has been returned to service prior to obtaining the duty shift superintendent (DSS) signature. (SER 93-076)

21. 2ICP-11.462, (Major), Functional Test of Pressurizer High and Low Pressure Loops Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

2ICP-11.462-1, (Minor), Functional Test of Pressurizer High and Low Pressure Loops Following As-Built Wire Tracing in 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures include a technical review including Technical Specifications, FSAR, associated drawings and CHAMPS. The tests are not a complete functional test of pressurizer heaters control instrumentation but only of circuits connected through 2C158. The tests are performed during cold or refueling shutdown conditions. Prior to performance, either the reactor coolant system (RCS) is open or vented to atmosphere and high pressurizer power-operated relief valve (PORV) actuation is disabled to meet low temperature overpressure protection (LTOP) requirements. Operation of the PORVs are not required during these conditions. If the PORVs affected have not been disabled prior to performance of this test, the RCS must be open or vented to atmosphere.

Adequate safety and control functions are maintained during testing. Steps were added listing conditions necessary to verify the equipment was returned to service prior to obtaining the duty shift superintendent (DSS) signature. (SER 93-075)

22. 2ICP-11.470, (Major), As-Built Wire Tracing of Miscellaneous Relay Cabinet 2C158, Revision 0. (New Procedure)

2ICP-11.470-1, (Minor), As-Built Wire Tracing of Miscellaneous Relay Cabinet 2C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures address as-built wire tracing and functional testing for circuitry in and through Unit 2 miscellaneous relay rack, 2C158.

Nuclear instrumentation system (NIS) circuitry affected by wire tracing includes the source range high flux at shutdown containment horn. A reactivity insertion accident resulting in an actuation of the source range high flux at shutdown containment horn are precluded by performing these procedures with no refueling operations.

Wire tracing also covers the automatic start circuitry for component cooling (CC) pumps on low header pressure. Manual control of both CC pumps is not affected by wire tracing precluding a reactor coolant temperature increase resulting from decay heat. If the CC pump auto start circuit on low header pressure would be disconnected during wire tracing, pump operation would continue with only the auto start feature affected. Circuitry traced cannot stop any running pump.

Wire tracing of circuitry for residual heat removal (RHR) return and suction valve permissives cannot shut any open valve. Only permissives to open shut valves are affected. Wire tracing will proceed only while purification flow has been established via the RHR system.

Automatic closing of the steam generator blowdown isolation valves is also affected. These valves are closed during shutdown and maintained in a closed condition during wire tracing. Wire tracing of this circuitry cannot open any of these valves.

Functional testing and independent verification of affected circuitry at the completion of wire tracing was performed. Steps were added to verify that testing of all circuits was been completed. (SER 93-030-01)

23. 1ICP-11.471, Unit 1, (Major), As-Built Wire Tracing of Miscellaneous Relay Cabinet 1C158, Revision 0. (New Procedure)

1ICP-11.471-1, Unit 1, (Minor), As-Built Wire Tracing of Miscellaneous Relay Cabinet 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedures address the as-built wire tracing and functional testing for circuitry in and through Unit 1 miscellaneous relay rack 1C158.

NIS circuitry affected by wire tracing includes the source range high flux at shutdown containment horn. A reactivity insertion accident resulting in an actuation of the source range high flux at shutdown containment horn are precluded by performing these procedures with no refueling operations.

Wire tracing also covers the automatic start circuitry for component cooling (CC) pumps on low header pressure. Manual control of both CC pumps is not affected by wire tracing precluding a reactor coolant temperature increase resulting from decay heat. If the CC pump auto start circuit on low header pressure would be disconnected during wire tracing, pump operation would continue with only the auto start feature affected. Circuitry traced cannot stop running pump.

Wire tracing of circuitry for RHR return and suction valve permissives cannot shut any open valve. Only permissives to open shut valves are affected. Wire tracing proceeds only while purification flow has been established via the RHR system.

Automatic closing of the steam generator blowdown isolation valves is also affected. These valves are shut during shutdown and maintained in a shut condition during wire tracing. Wire tracing of this circuitry cannot open any of these valves.

Functional testing and independent verification of affected circuitry at the completion of wire tracing was performed. Steps were added to the major procedure verifying that testing of all circuits was completed.

The margin of safety is not reduced since adequate safety and control functions are maintained during wire tracing and ensured after wire tracing is completed. (SER 93-030)

24. 1ICP-11.472, Unit 1, (Major), Functional Test of Nuclear Instrumentation Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1ICP-11.472-1, Unit 1, (Minor), Functional Test of Nuclear Instrumentation Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review was performed including Technical Specifications, FSAR, associated drawings and CHAMPS.
The tests are not a complete functional test of nuclear instrumentation (NI) but only of circuits connected through 1C158. All three ranges of NIs are included. Steps were included for setup prior to testing each NI range. Appropriate precautions were made not

to violate Technical Specifications during shutdown. Source range high flux at shutdown containment horn is disabled during source range testing. Prerequisite steps were included to ensure no refueling operations and at least one source range is operable before testing begins on a source range channel. Steps were included to switch audio count rate to the operating source range before testing begins. During power range testing automatic rod stops are disabled to facilitate circuit testing. A prerequisite of performance is no rod movement. Steps were added listing conditions necessary to verify equipment was returned to service prior to obtaining DSS signature in the major procedure. The test procedures do not result in a reduction in the margin of safety since adequate safety and control functions are maintained during testing. (SER 93-018)

25. 1ICP-11.473, Unit 1, (Major), Functional Test of Chemical and Volume Control Instrumentation Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1ICP-11.473-1, Unit 1, (Major), Functional Test of Chemical and Volume Control Instrumentation Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review was performed including Technical Specifications, FSAR, associated drawings and CHAMPS.

The tests are not a complete functional test of chemical and volume control (CVCS) instrumentation but only of circuits connected through 1C158. Relays in cabinet 1C158 are tested in these procedures by simulating volume control tank level transmitters 1LT-112 and 1LT-141. The functions of the relays are removed during testing by opening sliders present in control room panel 1C04. This provides for testing with an ohmmeter for the relay contacts and is outside 1C158. Independent verification is used when opening and closing sliders ensuring that tested equipment is returned to service. This test is performed during cold shutdown. A prerequisite of the volume control tank being bypassed prevents abnormal flow paths that could be generated if there was purification flow. These procedures do not alter the boric acid injection flow path required by TS 15.3.2.A. Steps were added listing conditions necessary to verify equipment is returned to service prior to obtaining DSS signature in the major procedure. (SER 93-009)

26. 1ICP-11.474, Unit 1, (Major), Functional Test of Tavg Loops for Condenser Steam Dump Control Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1ICP-11.474-1, Unit 1, (Major), Functional Test of Tavg Loops for Condenser Steam Dump Control Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review was performed including Technical Specifications, FSAR, associated drawings and CHAMPS.

The tests are not a complete functional test of Tavg instrumentation but only of condenser steam dump control circuits connected through 1C158. Miscellaneous cabinet 1C158 relays tested energize solenoids for controlling fast opening, closing and modulating of condenser steam dump valves, with Tavg, auto stop trip and turbine first stage pressure inputs. These relays also provide main steam reheat valves and a gas stripper control vale with an auto stop trip signal. The functions listed are removed during testing by fuse removals and opening sliders present in control room panel 1C03. This provides for convenient testing with an ohmmeter of relay contacts and is outside 1C158. Independent verification is used when opening/closing sliders and during the removal and installation of fuses. This test is performed during cold shutdown. A prerequisite condition of no turbine stop valve stroking is a requirement for test setup only. It is not a plant condition precautionary measure. Steps were added listing conditions necessary to verify equipment is returned to service prior to obtaining DSS signature in the major procedure. (SER 93-033)

27. 1ICP-11.476, Unit 1, (Major), Functional Test of RHR Valve Interlocks, Low Power Auto Rod Withdrawal Stop and CCW System Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1ICP-11.476-1, Unit 1, (Minor), Functional Test of RHR Valve Interlocks, Low Power Auto Rod Withdrawal Stop and CCW System Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review was performed including Technical Specifications, FSAR, associated drawings and CHAMPS.

The tests are not a complete functional test of component cooling (CC) pumps standby start feature on low header pressure but only circuits connected through 1C158. The functions of 1PC-639-X1 and 1PC-639-X2, automatic start of CC pumps on low header pressure are tested by manually stopping 1P11B, CC pump causing low header pressure and observing 1P11A, CC pump start and annunciation occurring. Relay 1PC-639-X2 is tested by observing the plunger pull in and drop out. The tests are performed during cold shutdown. 1P11B in service is a prerequisite. 1P11A starts as a result of the tests. Flow in CC system is checked during the tests.

Both CC pumps are maintained in service providing CC flow to the chemical volume and control system (CVCS) non-regenerative heat exchangers for residual heat removal. The margin of safety is not reduced since adequate safety and control functions are maintained during and ensured after testing. (SER 93-034)

28. 1ICP-1.477, Unit 1, (Major), Functional Test of Pressurizer Low Level and High Pressure Loops Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1IPC-11.477-1, Unit 1, (Minor), Functional Test of Pressurizer Low Level and High Pressure Loops Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A technical review of the Technical Specifications, FSAR, associated drawings and CHAMPS was performed.

The tests are not a complete functional test of pressurizer heaters control instrumentation but only that portion of the circuits connected within 1C158. 1C158 relays are tested in these procedures by simulating pressurizer level transmitters 1LT-427, 1LT-428 and pressure transmitter 1PT-431. The tests are performed when the reactor is in cold shutdown and depressurized, 1PC-427, the normal letdown isolation valve is shut during test but this valve is upstream of 1CV-371A which is shut when letdown flow for purification is provided by the RHR system during shutdown. Purification flow via RHR is not secured during testing and a flow path to the reactor core for boric acid injection is maintained during the tests. The tests do not cause or create any drainage flow paths of reactor coolant. Steps were included for testing automatic circuits for pressurizer heaters. A prerequisite for pressurizer heaters to be racked out and red tagged was added.

Functional testing of affected circuitry is performed at the completion of testing. Verification of instrument signal plug removals and opening of sliders are performed before testing begins and performed again when signal plugs are installed and sliders are closed ensuring the tested equipment is returned to service. Steps were added to the major procedure verifying that testing of all circuits was completed and lists conditions necessary to verify the equipment was returned to service prior to obtaining the DSS signature. (SER 93-035)

29. 1ICP-11.478, Unit 1, (Major), Functional Test of Pressurizer Pressure Loops Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

1ICP-11.478-1, Unit 1, (Minor), Functional Test of Pressurizer Pressure Loops Following As-Built Wire Tracing in 1C158, Revision 0. (New Procedure)

Summary of Safety Evaluation: A complete technical review including Technical Specifications, FSAR, associated drawings and CHAMPS with format upgrade using the writer's guide for maintenance procedures was performed.

The tests are not a complete functional test of pressurizer heater control instrumentation but only of circuits connected through 1C158. 1C158 relays are tested by simulating pressurizer pressure transmitters 1PT-429, 1PT-430, 1PT-431, 1PT-449 and 1PT-440. Relay contacts are isolated using selector switches specific to the affected valve. This provides for testing with an ohmmeter and is outside 1C158. This test is performed during cold or refueling shutdown. During performance, the reactor coolant system (RCS) is open to atmosphere with the vessel head removed. Operation of the PORVs tested is not required during these conditions. With the RCS open to atmosphere, low temperature overpressure protection requirements are met.

Adequate safety and control functions are maintained during testing. Steps were added listing conditions necessary to verify the equipment was returned to service prior to obtaining DSS signature. (SER 93-036)

30. IT-12, Unit 1, (Major), Inservice Testing of Component Cooling (CC) Pumps and Valves (Quarterly), Revision 5. (Permanent)

Summary of Safety Evaluation: The CC surge tank is normally vented to the atmosphere. The vent valve is automatically closed upon a high radiation signal from the CC pump suction header. The change shuts the surge tank vent valve during normal operation to maintain containment isolation and prevent an uncontrolled path to the environment for radioactivity. Installation of a pressure gauge on the CC surge tank is required to allow operators to monitor surge tank pressure.

Atmospheric pressure is not required for proper operation of the CC system. The existing surge tank relief valve maintains system pressure below the maximum allowable valve. Although the loss of CC is not an accident previously evaluated in the PBNP FSAR, the operation of the CC system is not affected.

The atmospheric vent valve must be shut for high radiation conditions caused by system in-leakage from the residual heat removal heat exchanger accident described in FSAR Section 9.3.3. CC is also required to be a closed system outside containment as a redundant containment isolation barrier. By maintaining the vent valve in the closed position, the closed system integrity is assured. The change decreases the consequences of a malfunction of the radiation monitor or the surge tank vent valve. The consequences of an equipment malfunction is not increased.

As CC operability is not affected, component availability is not reduced and there is no impact on the Basis for Technical Specification 15.3.3.c. The change increases the margin of safety by ensuring that the redundant containment isolation barrier remains intact. (SER 93-071)

31. IWP 92-090*A, Unit 2, (Minor), Removal of Safety Injection (SI) Switchover Logic, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedure removes the high head safety injection (HHSI) pump automatic switchover circuitry per MR 92-090. During installation of the modification, each train of Unit 2 safeguards control power is red tagged out of service to isolate the components located in the safeguards racks for removal. Only one train of Unit 2 safeguards control power is red tagged at a time. Unit 1 safeguards control power is not affected. Each valve is electrically isolated while changes are made to the valve control circuitry.

Since the modification only removes circuitry that is not required for the safe operation of the plant, no accident or malfunction can result. The continuity of wires installed to complete the daisy chain of safeguards control power is verified to ensure that no piece of equipment is left without power. The ORT 3 & 6 tests, along with the safeguards logic test, checks that all safeguards components are connected to safeguards control power. (SER 93-069)

32. OP-4A, (Major), Filling and Venting Reactor Coolant System, Revision 38. (Temporary)

Summary of Safety Evaluation: The low temperature overpressure setpoint for both units failed to take into account the differential pressure across the core with reactor coolant pumps (RCPs) running. This means the reactor vessel wall could experience higher pressures than the pressure instruments that actuate the LTOP relief valves. The error was calculated to be 63 psid for two pumps running and 25 psid for one pump running.

In the NRC LTOP system safety evaluation, a setpoint of 425 psig was approved based on:

- The use of zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions.
- The predicted maximum pressure transient is the sum of the overshoot magnitude and the setpoint magnitude.
- The worst case transient was a mass input transient. The calculated pressure overshoot was less than or equal to 94.5 psig.
- Only one PORV is assumed to open.
- No credit was taken for the RHR reliefs.
- The Appendix G limits were not exceeded for the worst case mass input transient.
- The heat input transient is less limiting than the mass input transient.

The current LTOP setpoint for both units is 415 psig. With this setpoint, and two RCPs running, the actual relief valve lift when the pressure on the vessel wall is 478 psig. With a maximum pressure overshoot of 94.5 psig, the maximum pressure that the vessel wall could experience is 572.5 psig. This pressure exceeds the zero degree heatup curve at approximately 151.5°F. If only one RCP was allowed to operate below this temperature, the maximum pressure the vessel wall could experience would be 534.5 psig. This is below the most limiting point on the zero degree heatup curve of 539 psig. Therefore, limiting RCP operation to one pump below 160°F prevents the zero degree heatup curve from being exceeded during the worst case mass input transient.

The changes involve only the restriction of limiting operation of reactor coolant pumps to a single pump at a time when the RCS temperature is less than 160°F. There are no expected effects except for the administrative controls placed to ensure only single pump operation. During fill and vent, both pumps are normally run together to check for the last remnants of air in the steam generator tubes. Instead, the pumps are operated singly as required to remove the large amounts of air prior to adding hydrazine for the final oxygen specification. The control switch for the RCP that is not operated is placing in the "pull-out" position and a red tag is placed on it to further restrict operation when the opposite pump is operated. TS 15.3.1.A requires only one RCP be operated or decay heat is removed using the RHR system. (SER 93-040)

33. Operations Special Order PBNP 93-004, (Minor), Reactor Coolant Pump Operation - Cold Shutdown, Revision 0. (New Procedure) The procedure limits reactor coolant pump operation when RCS cold leg temperature is <160°F in support of SER 93-040.

Summary of Safety Evaluation: The low temperature over-pressurization (LTOP) operating setpoint for the pressurizer power operated relief valves (PORV) appears to be non-conservative.

The PORV operating setpoint of 415 psig did not consider the actual or potential differential pressure created by reactor coolant pump (RCP) operation, or the difference in pressure at the reactor vessel versus the elevation of the pressure sensing instruments which actuate the PORVs. Because of those non-conservatisms, operation of two RCPs below approximately 152°F may allow the desired maximum pressure for the reactor vessel to be exceeded. Two RCP d/p is estimated at 63 psi.

Since the pressure at which the PORVs must operate is temperature dependent, we can reduce pressure at the most sensitive site (reactor vessel cold leg) by controlling RCP operation when the RCS cold leg temperature is below the limiting temperature. This condition applies to both units. (SER 93-027)

34. Operations Special Order PBNP 93-05, (Minor), Restoration of Control Room, Computer Room and Cable Spreading Rom HVAC, Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedure provides additional guidance for operators to restore control room ventilation and establish emergency filtration following SI actuation or 480 Vac undervoltage. PBNP 93-05 addresses identified design deficiencies.

Because the control room HVAC system is powered from predominantly non-safeguards sources that are locked-out on SI actuation or under voltage guidance is required for the restoration of HVAC until design modifications can be implemented to improved the redundancy of HVAC equipment and power supplies. (SER 93-038)

35. PBTP-006, (Major), Special Runout/Cavitation Test of 1P15B Safety Injection Pump, Revision 0. (New Procedure)

Summary of Safety Evaluation: PBTP-006 determines the performance of 1P15B safety injection (SI) pump when the pump is powered by G02 emergency diesel generator (EDG) at frequencies greater than 60 Hz. Two specific lineups are tested: cold leg injection plug core elude with suction from the refueling water storage tank (RWST), and cold leg injection with suction from the RWST.

The test is performed in response to condition report (CR) 91-481 which identifies SI pump cavitation during the performance of special maintenance procedure (SMP) 1082. In addition the testing addresses concerns in CR 92-644. CR 92-644 identifies a pump runout concern if an SI pump is lined up for cold leg and deluge injection.

The testing does not pose an unreviewed safety question because:

- The test is performed after the core is off-loaded during U1R20. The test is included and reviewed in the outage schedule.

- G02 EDG is out-of-service during the test under the limiting condition for operations in TS 15.3.7.B.1.G. G02 EDG is available to provide backup power to spent fuel pool (SFP) cooling pump P12B with a manual tie-in of the breakers if the pump is needed. Note that only one SFP cooling pump is required to remove decay heat with P12A planned to run during the test. G01 EDG is available during the test and is tested prior to G02 being declared out of service.

- Electrical loading of G02 EDG is not adversely affected because the diesel is run in the exercise mode. Thus, the EDG is not aligned to start and automatically accept load sequencing on an SI signal and undervoltage condition.

- Motor-operated valve 1SI-866B is used to control flow. The use of this valve as a throttling valve does not adversely affect the valve's function.

- Instrumentation is temporarily installed to record pump suction and discharge pressure, discharge flow, vibration and motor current during the test. Procedural temporary modification controls are utilized to control this test instrumentation. These controls assure the test instrumentation does not affect any structure system or component (SSC) and only records test data.

- The procedure precautions and limitations give the operators permission to secure from the test and return the electrical lineup to normal if needed at any time during the test.

- SI pump 1P15B and a limited amount of equipment powered from 1B04 are powered by the EDG at frequencies up to approximately 63 hertz during the test. A technical review was completed with no adverse effects expected from running these components at higher frequencies.

- The test sequence is written so the pump cavitation point is approached slowly by throttling open MOV 1SI-866B in small increments. The operators will observe for fluctuating discharge pressure, flow and motor current as signs of cavitation. In addition, a person is stationed at the pump to listen for cavitation. If cavitation begins to occur, the operator immediately throttles shut on 1SI-866B to stop the cavitation.

- The pump manufacturer was contacted. For short periods of cavitation (as may occur during the test), no adverse affects on the pump are expected. After the test and prior to return to criticality, the pump inservice test is performed to verify proper pump performance.

The above measures keep the revised configuration of the plant within initial conditions and assumptions of the FSAR. (SER 93-037)

36. PBTP-008, (Major), Wire Tracing and Functional Testing of Miscellaneous Breakers and Circuits in Main Control Boards 2C03, Revision 0. (New Procedure)

PBTP-008-1, (Minor), As-Built Wire Tracing of Section 2C03 Risers 46, 48, 50, 52, and 54 (Train A), Revision 0. (New Procedure)

PBTP-008-2, (Minor), As-Built Wire Tracing of Section 2C03 Risers 43, 45, 49, and 51 (Train B), Revision 0. (New Procedure)

PBTP-008-3, (Minor), Functional Test for Main Control Board Section 2C03 Risers 46, 48, 50, 52 and 54 and Associated Circuitry (Train A), Revision 0. (New Procedure)

PBTP-008-4, (Minor), Functional Test for Main Control Board Section 2C03 Risers 43, 45, 47, 49, and 51 and Associated Circuitry (Train B), Revision 0. (New Procedure)

Summary of Safety Evaluation: The as-built activities involve hand-over-hand wire tracing and followup testing for Unit 2 safety-related components in main control board 2C03. The components involved are those with wiring related to the Unit 2 residual heat removal system, Unit 2 main steam system, component cooling system, Unit 2 steam driven auxiliary feedwater system, and Unit 2 feedwater isolation.

Systems remain operable during wire tracing and are not affected unless a wire becomes disconnected. LCO entry is not required during routine wire tracing as this activity alone does not affect the operability of a system or component. However, LCO entry will occur if wire(s) become disconnected.

Other systems having wiring or components in section 2C03 of the main control board include Unit 2 lube oil, Unit 2 condensate and feedwater, circulating water, Unit 2 EH control, and Unit 2 steam dump. These systems are not directly affected as circuitry for these systems are not traced. However, if these systems have wiring in risers with safety-related circuitry that are traced, any wires which were inadvertently disconnected are found by visual inspections.

Upon completion of wire tracing for each train, followup testing is performed on all circuitry traced. This is done through functional and electrical tests as well as reliance on ORT-3, IT-09, IT-285, and IT-395. All testing maintains system operability. Also, all terminal blocks and components are visually inspected before and after wire tracing to ensure no wires have become disconnected during wire tracing. (SER 93-059)

37. PBTP-010, (Major), Wire Tracing and Functional Testing of Miscellaneous Breakers and Circuits in Main Control Board 2C04, Revision 0. (New Procedure)

PBTP-010-1, (Minor), As-Built Wire Tracing of Section 2C04 Risers 56, 58, 60, 62, and 64 (Train A), Revision 0. (New Procedure)

PBTP-010-2, (Minor), As-Built Wire Tracing of Section 2C04 Risers 53, 55, 57, 59, and 61 (Train B), Revision 0. (New Procedure)

PBTP-010-3, (Minor), Functional Test for Main Control Board Section 2C04 Risers 56, 58, 60, 62, and 64 and Associated Circuitry (Train A), Revision 0. (New Procedure)

PBTP-010-4, (Minor), Functional Test for Main Control Board Section 2C04 Risers 53, 55, 57, 59, and 61 and Associated Circuitry (Train B), Revision 0. (New Procedure)

Summary of Safety Evaluation: As-built work was performed in main control board 2C04. The work entailed visual inspections, tightening of connections, hand-over-hand wire tracing and testing of safety-related circuitry in 2C04. The systems with circuitry traced were Unit 2 reactor coolant system, chemical and volume control system, reactor protection system, nuclear instrumentation system, and containment purge supply and exhaust system.

The containment purge supply and exhaust system and the containment isolation system remained operable during the work. This was possible because of provisions set forth in PBTP-010 which required one train of these systems to be traced and tested before the other train was traced. However, if a wire was discovered disconnected, it was brought to the immediate attention of the duty shift superintendent and the responsible engineer. The wire's function and impact on the system was then analyzed. A limiting condition for operation (LCO) would be entered if necessary, and the wire was promptly installed.

Other systems having wiring or components in section 2C04 of the main control board includ~ Unit 2 vital instrument bus 120 Vac system, Unit 2 rod drive control system, Unit 2 control rod drive and cooling, heating and ventilation system, containment cleanup system heating and ventilation, and reactor cavity cooling heating and ventilation system. These systems were not directly affected as circuitry for these systems was not traced. Furthermore, the majority of these components are physically separated from those components traced. However, if the systems had wiring in risers with safety-related circuitry traced, any wires which were inadvertently disconnected were discovered by visual inspections which the electricians were required to perform per PBTP-010.

Upon completion of wire tracing for each train, followup testing was performed on circuitry traced. Interim conditions during testing were reviewed and found not to be a safety concern. Testing was accomplished by utilizing functional and electrical tests as well as reliance on ORT-3, IT-365, ICP 2.17, and ICP 2.18. Testing maintains system operability. Also, terminal blocks and components were visually inspected before and after wire tracing to ensure that no wires became disconnected during wire tracing. (SER 93-060)

38. PBTP-011, (Minor), Control Room, Computer Room, and Cable Spreading Room Heating, Ventilation and Air Conditioning (HVAC) Measurements, Revision 0. (New Procedure)

Summary of Safety Evaluation: The test is a modified TS-9, "Control Room Heating and Ventilation System Monthly Checks," test to perform the following:

- Measure control and computer room pressures and airflows with computer room dampers open during operational Modes 3, 4, and Modes 3 and 4 combined.

- Temporarily adjust control room supply airflows to determine airflow distribution with computer room dampers open during operation Modes 3 and 4 combined, while maintaining +1/8" water pressure in the control room and computer room.

The control room HVAC system cannot initiate an accident which would result in a radiological release to the environment, and the test procedure does not change the operation of the system in any manner such that it might initiate an accident. The system is operated in design modes (with the exception that during a portion of the test the control room and computer room will, for purposes of recirculation, be one volume) for data collection and air flow distribution testing. Required control room overpressure and cooling is maintained throughout the test.

TS 15.3.12 states that the emergency filtration system must always be available while operating or refueling. This test does not affect the Basis of this Technical Specification since the emergency filtration system will be available. (SER 93-074)

39. PBTP-018, (Major), Wire Tracing and Functional Testing of Miscellaneous Breakers and Circuits in Main Control Board C01, Revision 0. (New Procedure)

PBTP-018-1, (Minor), As-Built Wire Tracing of Main Control Board Section C01 Riser 26 and Riser 28B0 (Train B), Revision 0. (New Procedure)

PBTP-018-2, (Minor), As-Built Wire Tracing of Main Control Board Section C01 Riser 25 and Riser 27B0 (Train A), Revision 0. (New Procedure)

PBTP-018-3, (Minor), Functional Test for Main Control Board C01 Riser 25 and Associated Circuitry (Train A), Revision 0. (New Procedure)

PBTP-018-4, (Minor), Functional Test for Main Control Board C01 Riser 26 and Associated Circuitry (Train B), Revision 0. (New Procedure)

PBTP-018-5, (Minor), Functional Test for Main Control Board C01 Riser 27B0 and Associated Circuitry (Train A), Revision 0. (New Procedure)

PBTP-018-6, (Minor), Functional Test for Main Control Board C01 Riser 28B0 and Associated Circuitry (Train B), Revision 0. (New Procedure)

Summary of Safety Evaluation: The as-built activities involved hand-over-hand wire tracing and follow up testing for safety-related components in main control board C01. The components involved were those with wiring in risers 25, 26, 27BO and 28BO. The affected systems were Unit 2 safety injection, containment spray, containment isolation, air recirculation fan coolers, service water, main steam, reactor coolant and fuel oil.

Systems remained operable during wire tracing and were not affected unless a wire became disconnected. Limiting condition for operation (LCO) entry is not required during routine wire tracing as this activity alone does not affect the operability of a system or component. However, LCO entry would occur if wire(s) become disconnected.

Other systems having wiring or components in section C01 included Unit 1 turbine generator, motor-driven auxiliary feedwater, fire pumps, instrument air, reactor makeup water, and Unit 1 safeguards. These systems were not impacted by the activities as their wiring was not traced and is physically separated from risers 25, 26, 27BO and 28BO.

Upon completion of wire tracing for each train, followup testing was performed on circuitry traced. This is done through functional and electrical tests as well as reliance on ORT-3, ORT-6, and IT-14. Testing maintains system operability. Also, terminal blocks on risers 25, 26, 27BO and 28BO and Unit 2 safety-related components were visually inspected before and after wire tracing to ensure that no wires had become disconnected during wire tracing. (SER 93-058)

40. PBTP-019, Unit 2, (Minor), Gravity Drain Test from the Refueling Water Storage Tank (RWST) to the Reactor Coolant System (RCS), Revision 0. (New Procedure)

Summary of Safety Evaluation: The procedure evaluates the effectiveness of using gravity drain as an alternate method of decay heat removal when AC power has been lost to a shutdown unit. Shutdown conditions include the RCS being open for refueling or maintenance or partially drained. In these conditions, the steam generators may not be available for decay heat removal. Gravity drain provides a method to maintain inventory to keep the core covered while boil off (latent heat) removes decay heat from the core.

Once adequate flows were verified using gravity drain, the appropriate steps were incorporated into SEP-3.0, "Loss of All AC Power to a Shutdown Unit." This activity was scheduled during unloaded core conditions to ensure no unreviewed safety questions were proposed by the activity. (SER 93-079)

41. RMP 56, (Minor), Calibration and Testing of Safety-Related Protective Relays Under Technical Specifications, Revision 7. (Temporary)

RMP 65, (Minor), Calibration and Testing of Safety-Related Protective Relays Under Technical Specifications, Revision 6. (Temporary)

Summary of Safety Evaluation: The change deletes testing and calibration steps except those required to calibrate and test the degraded gird voltage relays for both units. The degraded grid relays require recalibration due to a recent setpoint change. The setpoint change, and the installation of the change, are controlled by procedure.

The relay setpoint was adjusted and verified during setting. Return to service testing consisted of verifying actuation of the auxiliary time delay relay when the degraded grid voltage relay was actuated by observing control room annunciators. The timing of the time delay relay was not affected by the setpoint change, and therefore was not required to be retested. The time delay associated with the under voltage (UV) relay itself was verified when calibrated.

During the time a degraded grid voltage relay is removed from its case, it is not available to perform its function. However, this is acceptable, since none of the relays are considered operable. The plant is in a 3-hour LCO, with a two week waiver of compliance, during performance of this procedure. Realizing, however, that even with the existing setpoint, the degraded grid voltage relays could perform some safety function, only one relay is removed from its case at a time. The operability of the other two relays on a given bus is verified before removing the third relay from its case.

Removing a relay from its case does not cause automatic functions. When relays are tested, they are tested one at a time. Two of three relay actuations are required to initiate automatic bus transfer. Thus, no automatic actuations are expected during performance of this procedure. (SER 93-002)

42. 1RMP 9133, (Major), Turbine-Driven Auxiliary Feedwater Pump Start on Bus A01 and A02 Undervoltage Refuel Calibration, Revision 0. (New Procedure)

1RMP 9133-1, (Minor), Turbine-Driven Auxiliary Feedwater Pump Start on Bus A01 and A02 Undervoltage Refuel Calibration, Revision 0. (New Procedure)

2RMP 9133, (Major), Turbine-Driven Auxiliary Feedwater Pump Start on Bus A01 and A02 Undervoltage Refuel Calibration, Revision 0. (New Procedure)

2RMP 9133-1, (Minor), Turbine-Driven Auxiliary Feedwater Pump Start on Bus A01 and A02 Undervoltage Refuel Calibration, Revision 0. (New Procedure)

Summary of Safety Evaluation: The upgrade included a technical review of Technical Specifications, FSAR, CHAMPS, electrical and logic drawings, and discussions with craft and supervisory persons.

While this specific surveillance test is not described in the FSAR, it is a TS required activity as shown in Table 15.4.1-1. The test methodology is similar to other ESF test methodology as discussed on TS Page 15.3.5-5. Test methodology is not changed from previous performances of this surveillance activity. The procedure is performed with the reactor in cold or refueling shutdown. The tested relays are tripped and restored to service prior to closing out the procedure. (SER 93-057)

43. SEP-2, (Major), Shutdown LOCA Analysis, Revision 0. (New Procedure)

SEP-2.1, (Major), Shutdown LOCA, Revision 0. (New Procedure)

SEP-2.3, (Major), Cold Shutdown LOCA, Revision 0. (New Procedure)

SEP-2.4, (Major), LOCA in a Drained Down Condition, Revision 0. (New Procedure)

SEP-2.5, (Major), LOCA with the Reactor Vessel Head Removed, Revision 0. (New Procedure)

Summary of Safety Evaluation: The new procedures address shutdown loss of coolant accidents (LOCAs). Shutdown LOCAs are large leaks of primary coolant greater than the capacity of charging to provide makeup. The LOCAs occur in the time frame after the accumulators are isolated in operating procedure OP-3C, until they are returned to service via OP-1A. This timeframe encompasses cold shutdown and refueling shutdown conditions. As such, these SEPs address a time frame that is for the most part outside of the FSAR licensing basis. During this time frame, systems may be out of service for maintenance, in a different lineup than for normal operations (e.g. cross-connected, RHR lineup, etc.). The procedures are designed to guide the operator through actions to protect the reactor core by maintaining core cooling and reactivity control as well as attempting to isolate the leak.

The procedures provide for appropriate notifications and implementation of EPIP 1.1. The procedures have leak isolation, containment isolation, and establishment of long-term cooling in common. They provide guidance to the operators on mitigation of beyond design basis accidents. Prior to these procedures, only operator experience was relied upon for mitigation. Operator experience is still heavily relied upon due to the variety and complexity of all system configurations possible during shutdown conditions. The SEPs are designed to assist the operator in achieving effective mitigation.

The procedures (with exception of SEP-2.5) were run on the simulator under several different circumstances. Problems discovered were resolved and corrected and possible future changes identified. SEP-2.5 was not run on the simulator as its configuration with the reactor vessel head removed could not be modeled. (SER 93-027)

44. SEP-3, (Major), Loss of AC Power to a Shutdown Unit, Revision 0. (New Procedure)

Summary of Safety Evaluation: SEP-3 provides guidance in the event of a loss of AC power during shutdown plant conditions. As such, the SEP addresses a timeframe that is for the most part outside of our FSAR licensing basis. During this timeframe, systems may be out of service for maintenance, in a different lineup than for normal operations (e.g. cross-connected, RHR lineup, etc.). The SEP guides the operator through actions to protect the reactor core by maintaining core cooling and reactivity control, utilizing the alternate shutdown panel if necessary as well as attempting to restore AC power.

The procedure provides for appropriate notifications and implementation of EPIP 1.1. It also provides directions to maintain core cooling and level while providing alternate methods to restore power to vital equipment. It provides guidance to the operators on mitigation of beyond design basis accidents. Prior to the procedure, only operator experience was relied upon for mitigation. Operator experience is still heavily relied upon due to the variety and complexity of possible system configurations during shutdown conditions. The procedure was written to assist the operator in achieving effective mitigation. (SER 93-028)

45. SMP 1088, Unit 2, (Minor), Safety Injection (SI) Pump P-15A Motor Connection Inspection, Revision 0. (New Procedure)

SMP 1089, Unit 2, (Minor), Safety Injection (SI) Pump P-15B Motor Connection Inspection, Revision 0. (New Procedure)

Summary of Safety Evaluation: The safety implications of SMP 1088 and 1089 are: The adequacy of the feeder cable and motor lead lugs and bolting hardware; and, the environmental qualification of the method of reinsulating the feeder cable for electrical stress control and the method of reinsulating the motor connection.

The evollutions do not affect the intended design, operation, function, or method of function of the SI pump motors. The environmental qualification status of the motor connections are maintained or improved. (SER 93-072)

46. SMPs 1132, 1132-1, 1132-2, 1132-3, 1132-4, 1132-5, 1132-6, MWRs 930909, 930910, 930911, 930912, and SER 93-007, As-Built Wire Tracing and Follow-up Testing in Section C01 of the Main Control Board. Hand-over-hand wire tracing of safety-related circuitry having wiring in Risers 21, 22, 23N, and 24N was performed. These risers contain primarily Unit 1 safeguards circuitry along with wiring for service water, motor-driven auxiliary feedwater, fire pumps, and Unit 1 turbine generator. Risers 21 and 23N are used for Train A wiring while risers 22 and 24N are used for Train B. Wires are traced one train at a time from safety-related components with wiring in these risers.

Summary of Safety Evaluation: Systems remained operable during wire tracing and were not affected unless a wire became disconnected. LCO entry is not required during routine wire tracing as this activity alone does not affect the operability of a system, structure or component. However, LCO entry would occur if wire(s) become disconnected.

Additional safety precautions were taken throughout the activities to prevent other possible system impact. Ceramic wire cutters were used wherever possible to prevent electrical shorts, and wire tie scraps were collected and discarded to prevent their interference with relay contacts.

Other systems having wiring or components in section C01 included fuel oil delivery, instrument air, reactor makeup water, and Unit 2 safeguards. These systems were not impacted by the activities as their wiring was not traced and was physically separated from Risers 21, 22, 23N, and 24N.

Upon completion of wire tracing for each train, follow up testing was performed on circuitry traced. This was accomplished through functional and electrical tests in SMP 1132 as well as reliance on ORT-3 and ORT-6. Testing maintains system operability. Also, terminal blocks on Risers 21, 22, 23N, and 24N were visually inspected before and after wire tracing to ensure no other wires for non-safety related equipment had become disconnected during wire tracing. (SER 93-021)

47. SMP 1139, Isolating and Replacing SW-4, Revision 0. (New Procedure)

Summary of Safety Evaluation: SW-4, the south service water zurn strainer outlet valve, was replaced with a rebuilt valve. The rebuilt valve was removed from the SW-9 position per SMP 1136 and replaced with a spoolpiece.

The SW pumps are safe shutdown equipment required to respond to a "worst case" fire for 10 CFR Appendix R concerns. Due to separation concerns resulting from the location of the particular SW pumps being removed from service, twice per shift fire watches are posted in the pump house and G02 EDG room as a prudent action to be consistent with Standing Order PBNP 4.12.7. The LCO is entered in accordance with PBNP 3.4.21, "Voluntary Entry into a Limiting Condition for Operation."

If an SI actuation signal is received while the three south pumps are isolated, then the affected unit turbine building feeder is isolated resulting in loss of coolant to the lube oil cooler which could damage turbine bearings. To prevent this damage, contingencies are included in the SMP to supply cooling to the lube oil cooler of the affected unit using fire water if there is an SI actuation signal. This is considered an emergency condition, therefore allowing the use of the fire water system for this purpose.

If the repaired valve cannot be installed within the 24-hour LCO or service water (SW) is needed to be routed to the south header from the east header, blind flanges can be installed on the header flanges and flow can be routed around the south zurn strainer bypass.

The SW system remains operable throughout the valve replacement and no other equipment important to safety is affected. The operable SW pumps are not affected because the isolated section of header is shown to be adequately leak tight prior to removing the valve. Adequate drainage is also provided in the pumphouse to ensure water draining from the SW pipe during valve replacement does not effect any other equipment or cause a personnel hazard. (SER 93-054)

48. SMP 1141, (Minor), Isolating and Replacing SW-2891, Revision 0. (New Procedure)

Summary of Safety Evaluation: SW-2891, south to north header crossconnect valve, was replaced with a rebuilt valve. The rebuilt valve was removed from the SW-4 position per SMP 1139 and replaced with another rebuilt valve.

The preliminary condition of having six SW pumps operable and SW-2890 shut and inoperable while not being in a LCO is supported by Calculation N-93-047 using the approved computer model of the SW system. The calculation demonstrates that three SW pumps have sufficient flow during a Unit 2 LOCA along with G02 failing to start, the most limiting condition.

The SW system remains operable, with regard to seismic concerns, throughout valve replacement since shims have been placed under each of the six discharge check valves to reduce loads on the header. The work is being done under a LCO which relaxes the single failure criteria for the SW system. Therefore, an additional failure of the SW system and it's potential effect on equipment important to safety is not considered. The operable SW pumps are not affected because the isolated section of header is shown to be adequately leak tight prior to removing the valve. Furthermore, flooding in the pumphouse does not affect the other SW pumps or diesel fire pump because adequate drainage is established in the pumphouse as determined during past SW valve removal. Flooding is evaluated further by the evolution coordinator during the performance of SMP 1141. Plastic is also placed over the P32C motor to protect it from spraying water when SW-2891 is removed. The only other equipment important to safety which could be affected are the circulating water pumps and traveling water screens. If leakage is excessive and SW-42 must be shut, routing water from the screen wash strainer blowdown to the Unit 1 circulating pumps before shutting SW-42 allows the circulating pumps to remain operable throughout the evolution.

The SW pumps are safe shutdown equipment required to respond to a "worst case" fire for 10 CFR Appendix R concerns. Due to separation concerns resulting from the location of the particular SW pumps being removed from service, twice per shift fire watches are posted in the pumphouse and G02 EDG room. (SER 93-077)

49. SMP 1143, (Minor), Isolating and Replacing SW-2890, Revision 0. (New Procedure)

Summary of Safety Evaluation: SW-2890, north to south header crossconnect valve, was replaced with a rebuilt valve. The rebuilt valve was removed from the SW-2891 position per SMP 1141 and replaced with another rebuilt valve.

During portions of the SMP, SW-2890 or SW-2891 is inoperable shut and not able to be opened due to the operator being removed or the valve tagged shut. Inoperability of either valve is defined as it not having the ability to be cycled remotely or manually. During these portions of the SMP, Technical Specification 15.3.3.D-1.b statement "All necessary valves, interlock and piping required for the functioning of the service water system during accident conditions are also operable.", does not apply with this valve closed provided all six SW pumps are operable. Calculation N-93-047 provides documentation showing that, in the most limiting condition (Unit 2 LOCA and G02 failing to start), adequate flow is supplied to the Unit 2 fan coolers for post-accident conditions via three SW pumps. In the event that less than six SW pumps are operable, a 24-hour limiting condition for operation (LCO) will be entered.

The SW system remains operable, with regard to seismic concerns, throughout the valve replacement since shims were placed under each of the six discharge check valves to reduce loads on the header. The work is done under an LCO which relaxes the single failure criteria for the SW system, therefore an additional failure of the SW system and its potential effect on equipment important to safety is not considered. The operable SW pumps are not affected because the isolated section of header is shown to be adequately leak tight prior to removing the valve.

The SW system is required to mitigate the radiological consequences of an accident. By removing three SW pumps from service when the north half of the east header is isolated, the systems ability to perform it's design function needed to be evaluated. This condition results in P32A, P32B and P32C remaining in service. Pumps A and B are powered from Train A and pump C is powered from Train B. Although, the PBNP Technical Specifications allow three pumps to be taken out of service under an LCO, the conservative interpretation is that one train should remain operable. This results in P32A and P32B, powered from Train A, being the operable train of SW pumps and P32C, Train B, being the train degraded under the LCO.

The SW pumps are safe shutdown equipment required to respond to a "worst case" fire for 10 CFR Appendix R concerns. Due to separation concerns resulting from the location of the particular SW pumps being removed from service, twice per shift fire watches are posted in the pumphouse and G01 EDG room. (SER 93-077-01)

50. STPT 1.2, Unit 2, (Major), Reactor Trip OTΔT, Revision 5. (Permanent)

STPT 1.3, Unit 2, (Major), Reactor Trip OPΔT, Revision 6. (Permanent)

Summary of Safety Evaluation: TS Change Request 160 reduces the minimum measured flow rate of Unit 2 by 2600 gpm. The reduction is necessary due to decreased flow rates resulting from steam generator tube plugging. Core thermal safety limits must be changed to allow for the lower flow limits. To ensure protection against departure from nucleate boiling (DNB), a change to the overtemperature and overpower ΔT (OTΔT and OPΔT) setpoints is required. For the new safety limit, the setpoints are to be changed such that they are moved in the conservative, more operationally restrictive direction. The change to the setpoint reduces the ΔT factor from 573.9°F to 570.0°F. This reduces the OTΔT setpoint by ≈4.3°F and the OPΔT setpoint by ≈0.264°F.

The more restrictive OTΔT setpoint provides less margin to the turbine runback and trip functions. However, normal operation of the unit is not affected and the new OTΔT setpoints will continue to provide enough margin for the typical delta flux penalty (1.22°F) experienced during a reactor startup early in core life.

The change is well within the range of the installed instruments. The setpoints are changed in accordance with accepted calibration procedures. No hardware changes are required to alter the setpoints. (SER 88-141-05)

51. STPT 1.2, Unit 2, (Major), Reactor Trip OTΔT, Revision 5. (Permanent)

STPT 1.3, Unit 2, (Major), Reactor Trip OPΔT, Revision 6. (Permanent)

STPT 4.1, Unit 2, (Major), Rod Stop and Turbine Runback Setpoints: Rod Stops, Revision 5. (Permanent)

STPT 4.2, Unit 2, (Major), Rod Stop and Turbine Runback Setpoints: Turbine Runbacks, Revision 4. (Permanent)

Summary of Safety Evaluation: This evaluation supersedes SER 88-141-05. The more restrictive overtemperature ΔT setpoint provides less margin to the turbine runback and trip functions. The turbine runback function is currently set to occur 3°F before an OTΔT reactor trip and 1.5 °F before an OPΔT reactor trip. The turbine runback function is provided to reduce power prior to receiving a trip from the OTΔT and OPΔT setpoints. This is an anticipatory control function only and is not considered in the accident analysis as a mitigating action.

The additional conservatism contained in the bistable for OPΔT is removed since it is no longer required. This conservatism was placed in the bistable to address problems with the f(Tave) Impulse units (TM-401O, 402O, 403O, and 404O). These problems were repaired by performing work under SPEED 87-03 and are tested annually to ensure no recurrence. The additional conservatism is no longer required.

Turbine runback setpoints will be adjusted such that: OTΔT Turbine Runback Setpoint = ΔTsp^1 - 1.0 °F; OPΔT Turbine Runback Setpoint = ΔTsp^2 - 1.0 °F

The new setpoints for a turbine runback ensure that a runback still occurs prior to a reactor trip but does not restrict unit operation. (SER 88-141-06)

52. STPT 21.1, Degraded Grid Voltage Relays, Sheets 74, 75, 75, and 77.

Summary of Safety Evaluation: Based on review of a calculation, the setpoints of the degraded voltage relays on 4160 V safeguards buses 1A05, 1A06, 2A05, and 2A06 were found to be non-conservative. Under certain conditions this could result in several safety-related loads having voltage levels lower than the minimum value required. On January 7, 1993, a Notice of Enforcement Discretion was granted by the NRC and certain compensatory actions were taken. The setpoint changes assure adequate voltage levels are present at all safety-related loads and to prevent the spurious stripping of offsite power from these buses and subsequent challenges to safety systems. The setpoint values are based on calculation N-93-002.

Operation of safeguards equipment for extended periods of time in a degraded voltage condition could result in the failure of the equipment to perform its safety function and mitigate the consequences of an accident. The new, higher setpoint for the degraded voltage relays allows the removal of offsite power and transfer to the EDGs at an appropriate voltage level. This prevents the operation of safeguards equipment in a sustained degraded voltage condition. The new setpoints are at the minimum required voltage. However, the setpoints could not be adjusted high enough to account for relay, calibration and potential transformer tolerances. This avoids setting the degraded voltage relays too high which may cause spurious stripping of offsite power under normal expected operating grid voltages.

Not allowing for inaccuracies potentially could result in the operation of the 4160 V and 480 V buses and equipment at levels slightly below that recommended. A number of factors justify the acceptability of this condition and include:

- Calculation N-93-002, which determined the new setpoints, is based upon worst case conditions and contains several assumptions which result in the calculation being conservative.

- Three relays are installed on each 4160 V bus. Any two of the relays sensing an undervoltage results in the separation of the bus from offsite power (2/3 logic).

- The 4160 V buses protected by these relays provide power to redundant safety-related loads. Thus, failure of the relays on one bus to operate at the exact setpoint voltage would subject only one safeguards train to voltages slightly less than the minimum required.

- For consistency all the relays are set at 113.1 V. However, this high of a setting is only necessary for the relays on the 1A05 bus. This means that appropriate voltage continues to be provided to the equipment supplied by the other buses at this setpoint and slightly below.

- Per conversations with the relay manufacturer, the expected tolerance is +/-0.4% over a range of control power from 100 Vdc to 140 Vdc. The control voltage for these relays is 125 Vdc supplied from station batteries and is not expected to vary much even under the postulated accident conditions.

- Per discussions with the relay calibrators, there was no observed drift in the relay settings since they were installed in the early 1980s.

- Once implemented, operations personnel will be alerted that the degraded voltage relay setpoints are not at the optimum level. Until actual voltage measurements are taken and any needed modifications to reduce the voltage drop to affected equipment performed, this order remains in effect. Control room personnel are instructed to increase surveillance of voltage levels once a notification of low grid voltage is received from system control or during transient conditions, and to manually initiate action in response to a degraded voltage condition prior to voltage reaching a level at which equipment damage could occur.

While the new degraded voltage relay setpoints give a high degree of confidence that the minimum voltage levels are maintained, the possibility exists that some safety-related equipment could operate at a level slightly below that required. The above listed factors illustrate why this possibility is remote and compensate for the potential condition.

Degraded bus voltage is not an initiating event for any accident or malfunction of equipment previously analyzed in the FSAR. The setpoint changes are made to ensure that safety-related equipment will be able to function at proper voltage levels. The setpoints are at a level which do not cause spurious stripping of offsite power and result in unnecessary challenges to plant safety systems and potentially a unit trip.

The degraded voltage relays are not discussed in the basis for TS 15.3.5. However, they are required to ensure engineered safeguards equipment remains available during degraded voltage conditions. These setpoint changes do not decrease the availability of this equipment to perform its function. (SER 90-059-01)

53. TS-33/TS-34, Surveillance Testing: Containment Accident Recirculation Fan-Cooler Units, Revision 10/13. (Permanent)

Summary of Safety Evaluation: During normal operation, this revision does not affect the service water system or any of the components which it supplies. The individual throttle valves for service water from the containment fan coolers is opened further, but this is offset because the combined throttle valve is throttled more.

During a safety injection, SW-2907 and SW-2908 are open. These valves bypass the combined throttle valve. Therefore, the more throttled position of this valve has no effect on containment fan cooler service water flow.

With SW-2907 or SW-2908 open, service water flow for each containment fan cooler is raised by approximately 100 to 150 gpm per fan cooler unit. This causes a slight reduction in service water header pressure with a resultant reduction in flow to other components supplied by service water. This reduction in service water flow is not sufficient to cause any detrimental effects to the equipment involved.

When transferring to containment sump recirculation during an accident, the service water throttle valves for the component cooling heat exchangers are opened until a differential pressure of 4 psid is achieved across the heat exchangers. Therefore, service water flow through the component cooling heat exchangers is the same as before. (SERs 91-024-02 and 91-024-03)

DESIGN CHANGES

The following modifications were installed prior to 1993 but were inadvertently omitted from previous reports:

1. MR 87-002*I, (Common), Fire Protection. MR 87-002*I installs a fire wall between the 1X04 and 2X04 low voltage station auxiliary transformers. The upgrade installs a 16'' high, 1' thick, and 21' long reinforced concrete wall on the existing X04 concrete foundation pad. The new fire wall is anchored to the existing foundation pad by coring holes and grouting dowels into the existing X04 foundation.

Summary of Safety Evaluation: The modification enhances the reliability of the preferred offsite power supply for the safeguards bus, the 1X04 and 2X04 low voltage station auxiliary transformers. This enhanced reliability is achieved by reducing the possibility of a fire in any one of the X04 transformers spreading to the other X04 transformer.

Low voltage station auxiliary transformers, 1X04 and 2X04, are not taken out of service or de-energized to perform the work outlined in MR 87-002*I. Loss of the X04 transformers is an analyzed safety question in the PBNP FSAR, as the diesel generators are provided for this purpose.

Construction work on the fire wall is prohibited during "reduced inventory." In addition, work on the fire wall is permitted only when transformers 1X03, 2X03, 1X04, 2X04 and G01 and G02 EDGs are in service. Precautions are taken to prevent bumping of the X04 transformers, coring into the wires and conduits cast in the existing concrete foundations, blowing out of the concrete forms, hoisting or suspending of any equipment, personnel, etc., above the X04 transformers, and collapse of the new fire wall because of insufficient curing of concrete prior to the removal of the forms. Furthermore, the concrete fire wall was designed to resist seismic loads. Therefore, operability of the safeguards bus and associated equipment important to safety is unaffected. (SER 87-022-02)

2. MR 87-034*H (Common), 480 V Motor Control Centers (MCCs). MR 87-034*H retrofits breakers 1B52-12C and 2B52-31C with amptector solid state overloads. These breakers are supply breakers for the "A" and "B" motor-driven AFP respectively.

Summary of Safety Evaluation: Calculation N-91-038 analyzed the current overload setpoints on selected 480 V safeguards motor protection. The setpoints on auxiliary feedwater pump (AFP) supply breakers (1B52-12C and 2B52-31C) were revised via MR 87-034*H.

Based on the calculation, the instantaneous pickup current should be increased from 2800 amps to 3500 amps. The increased pickup current allows a maximum asymmetrical offset on starting, assuring operability. The 2800 amp setting may not assure operability of the pump motor under starting conditions. (SER 88-137-03)

3. MRs 87-103*A, (Unit 1) and 87-104*A, (Unit 2), Secondary Sampling System. The modifications install on-line ion chromotographs in the Unit 1 and 2 turbine halls. This includes moving the steam generator routine sample point from the primary sample rooms to the feedwater sample panels. The drains go to the turbine hall sump.

Summary of Safety Evaluation: The physical description of the sample lines requires a FSAR change because the pressure of the steam generator sample lines are not reduced until after they leave the sample room and go to the feedwater sample panels in the turbine hall. The grab sample capability at the primary sample sink is not changed but the sample lines going to instrumentation rack RK-21 are routed out to the sample panels in the turbine hall. The line going to RE-219 remains. Pipe whip analyses on the 3/8" sample lines [illegible] not required (FSAR Appendix E).

FSAR Table I.2-1 has a value of 1080 gal/day for secondary system sampling. This value should be increased to the more reasonable 3410 gallons per day with the new sample panels. The steam generator samples and main steam samples add 905 gallons per day per unit and the on-line ion chromatograph adds 190 gallons per day per unit. The total sample volume after all the modifications are completed should be 5600 gallons per day (4 gpm) for both units for secondary sampling.

Addition of the steam generator samples to the existing feedwater panel do not adversely affect the temperature control of the feedwater panel. The initial design of the panel allowed for four additional sample heat exchangers to be installed in the sample bath. The steam generator samples are already cooled to <130°F by the component cooling sample coolers in the primary sample room.

The sample lines, including the extension of the steam generator sample lines are outside of the QA boundary for any system and the sample lines are classified as Seismic Class III per FSAR Design Criteria 1.3.

The only analyzed accident that could have any effects on the movement of the steam generator sample lines would be the steam generator tube rupture accident. The only difference in the accident is that now steam generator samples are being directed to the turbine hall sump as opposed to the blowdown tank. During a tube rupture, the secondary side of the plant will be contaminated due to other secondary system leakage throughout the turbine hall at a much higher flow than the proposed sampling system. The total curies released from the plant is the same or less, however the route the sample takes to the lake and thus the dose to the public is the same. In actuality, the dose to the public will be reduced because the shorter response time for RE-219, will cause blowdown and sampling to be secured quicker than before. The assumed flowrate to RE-219 was 900 cc/minute in previous evaluations of the radiation monitoring system (NEPB 87-203). The modification increases the flow from the steam generator to RE-219 to 1400 to 1500 cc/minute up to the point where RE-219 branches off of the sample line. The flow to RE-219 is then reduced to 400 cc/minute for the short distance to the monitor.

To address the seismic classification of the Unit 2 steam generator sample lines above the safe shutdown switchgear, other sample lines that were attached to the wall were removed to facilitate the installation of the switchgear. This modification replaces sample lines on the wall using the sample method originally used for the other sample lines. (SER 91-051)

4. MR 88-018*R (Unit 1), Main Control Boards. MR 88-018*R modifies the range of existing control room meters to resolve operator recommendations made during the control room design review project. The meters undergoing range changes are: 1T-421, pressurizer surge temperature; 1T-425, pressurizer liquid temperature; and F-4007/4014, auxiliary feedwater pump (AFP) flow.

   Summary of Safety Evaluation: MR 88-018*R changes the ranges of 1T-421 and 1T-424, to match the range of 1T-425 which was changed previously, so all three pressurizer temperature indicators have the same range. The MR also changes the ranges of F-4007 and F-4014 so the maximum design flow (200 gpm) does not equal the maximum flow that can be indicated on the meters.

   The modifications to the auxiliary feedwater (AFW) loops affect a prior documented technical commitment to the NRC regarding the AFW system. RG 1.97 recommends that AFW flow meters provide 0 to 110% design flow. In WE's response to the NRC, the ranges of the meters are given as 0 to 200 gpm, providing indication from 0 to 100% design flow. With this modification, the meters provide from 0 to 150% design flow. It should also be noted the ranges of T-424 and T-425 were included in WE's response to Regulatory Guide 1.97, even though no range recommendations were made for these meters. Shifting the ranges of these meters from 0-700°F to 50-750°F still allows monitoring of expected pressurizer temperatures.

   Only one AFW flow transmitter is taken out of service at a time. TS Table 15.3.5-5 allows one channel of AFW indication to be taken out of service for up to 48 hours. This applies to the combination of AFP discharge flow indicators and AFW flow to steam generator flow indicators. The indicators involved in this modification can remain out of service for longer periods of time provided AFW flow indication to each generator is available during this time. TS also allows a motor-drive AFP to be out of service for up to 7 days for one or two unit operation.

   No EOP setpoint changes are required. Computer software change is required as specified in the modification request. (SER 91-027-12)

5. MR 88-052, (Common), Fire Protection. The modification changes the fire pump test header to allow testing without additional equipment.

   Summary of Safety Evaluation: The existing configuration is a header on the west pump house wall. To perform the pump test, six 50' rubber hoses are connectors to the header. The rubber hoses' pressure drop is too high to allow the required 150% flow rate. The modification replaces the existing header with an integral header/playpipe assembly. The new assembly does not add additional piping runs to the system. The addition does not affect the normal operation of the fire pumps. Additional fixtures are Piping Class KB and meet existing pressure and temperature specifications. A vehicle bumper is included to prevent accidental damage by collision.

The only change to the FSAR is a possible change in the pictorial representation of the test header on Figure 9.6-1 depending on the final configuration of the header. (SER 88-073)

6. MR 90-075*A, (Unit 2), Condenser. The modification installs impingement grating which deflects turbine steam exhaust, thereby protecting the tubes from ste n erosion. The general location of the grating is near the top of the tube bundles and along the outer tube bundle perimeter.

Summary of Safety Evaluation: MR 90-075*A installs impingement grating that deflects turbine steam exhaust, and thereby protects the tubes from erosion. Installation requires that the plant be in a cold shut down condition. The grating is located where tube erosion has been identified by both eddy current and visual inspections.

The turbine exhaust deflects 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 such as KNPP.

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

7. MR 91-166*A and *B (Common), HVAC/Instrument Air. MR 91-166 provides inservice testing capabilites to various HVAC systems.

Summary of Safety Evaluation: MR 91-166*A addressed control room and cable spreading room chilled water system enhancements. This included pump suction and discharge pressure indication, flow indication, and FST capabilities for the chilled water TCV. The control room/cable spreading room chilled water systems are presently non-QA, non-safety-related systems. The MR treats these systems as essentially safety-related. These chilled water piping systems are presently non-seismic, and the changes are designed as such. Should these systems be upgraded in the future, a full seismic evaluation will be performed at that time.

MR 91-166*B addressed similar FST capabilities for TCVs on the residual heat removal, condensate and feedwater/safety injection, auxiliary feedwater area coolers. Each of these coolers has a bypass line in parallel with the TCV so cooling water flow can be maintained during the installation phase, and therefore the functionality of these systems are not affected by the changes. The worst case result of the "as modified" configuration is to lose air to these TCVs, at which point they would simply go to their fail-safe position.

The MRs change the instrument air lines for control room/cable spreading room chilled water TCVs. Should the activities degrade instrument air in C58 (control room/cable spreading room HVAC control cabinet), the dampers and fans for the control room emergency filtration could be disabled. The work however, is of short duration and precautions are taken to prevent the system from depressurizing. The system is easily recoverable within the 7-day LCO allowed by TS. (SER 92-109)

8. MR 92-009-02, (Unit 1), Main Steam. The modification removes sufficient material from inside of the MSIV packing gland follower so that it becomes physically impossible for the packing gland follower to come in contact with the MSIV shaft when installed. Inspection evidence following disassembly of 1MS-2017 and 1MS-2018 at the end of U1R19 shows that MSIV shaft galling had occurred due to contact between the MSIV shaft and packing gland follower. This galling was the primary cause of the failure of 1MS-2018 during stroke

testing at the end of U1R19. Modification to the packing gland followers prohibit MSIV shaft galling, therefore reducing the probability of valve malfunction.

Summary of Safety Evaluation: The packing gland followers for each main steam isolation valve (MSIV) have their inner diameters increased so it is physically impossible for the packing gland follower to become canted sufficiently to make contact with the MSIV shaft.

The function of the MSIV packing gland follower is solely to transmit pressure to the MSIV shaft packing. The modification does not affect this function, or the ability of the packing gland follower to perform this function in any manner whatsoever. Additionally, the function of the MSIV and its ability to perform that function is, likewise, not affected in any manner. (SER 92-025-03)

9. MR 92-144*C (Common), Component Cooling (CC) Water. MR 92-144*C installs the electrical scope, includes the activation of automatic closure of CCW-LW-63 and 64 and the interlocks of the radwaste components cooled by CC water. The interlocks close valves or deenergize radwaste equipment that may be damaged by loss of CC cooling.

MR 92-144*C consists of activating the containment isolation (CI) interlocks to the new safety-related circuit installed per MR 92-144*B that automatically places the CC system in a configuration that strictly meets the criteria in FSAR Appendix A for class break isolation. The design ensures closed loop operation of the CC system. This is accomplished by providing isolation of the Class III radwaste piping from the Class I CC system.

Summary of Safety Evaluation: The enhancement to the circuit provides for automatic closure of valves CCW-LW-63 and CCW-LW-64 on a CI signal from either train. It provides for automatic tripping of the associated radwaste equipment requiring CC flow supplied through these valves to prevent radwaste equipment damage. It also provides annunciation of the CC valve closure or valve auto closure feature defeated in the control room, tripping of CC cooled radwaste equipment, and it annunciates loss of CC initiated radwaste tripping in the radwaste equipment control area.

The modification is an enhancement to the system design increasing reliability and ensuring conformance to existing design criteria. (SER 92-094-02)

The following modifications were installed in 1993:

1. MRs 84-227, 84-227*B, 84-227*C (Unit 1), 84-228, 84-228*C, (Unit 2), Safety-Related Instrument Buses. The modifications install static transfer capability for the inverters on all four safety-related instrument buses. The static transfer switches transfer to the alternate AC source upon an inverter failure or a fault condition that causes instrument bus voltage to drop below a preset level. Following a transfer to the alternate source, the affected instrument buses are manually transferred back to the normal or swing inverter via a switch at each inverter or static transfer switch. The instrument bus alternate AC source is intended to be used only until the affected instrument buses can be returned to their normal or swing inverter supplies.

Summary of Safety Evaluation: The static transfer switches are seismically qualified and have the same ratings as their associated inverters. New raceways and equipment are seismically installed. Existing safety train separation is maintained.

The alternate AC source is non-safety-related. Equipment associated with the alternate source is non-QA. This is acceptable because a failure of the alternate source does not cause an inverter failure or prevent the inverters from supplying their respective instrument buses. The interface between the non-safety-related alternate source and the safety-related instrument buses is at the static transfer switches, which are safety-related. The alternate source transformer is equipped with surge and lightning protection to reduce the potential for voltage transients on the alternate source. The static switches have surge protection and design features which prevent voltage transients on the alternate source from affecting static switch operation.

Following testing of each of the Unit 2 static transfer switches, the associated Unit 2 instrument buses are transferred to the alternate source to verify proper operation. This is performed during a refueling outage to minimize risk. (SER 91-083-02)

2. MR 87-016*D, (Common), Computer. MR 87-016*D installs a data link between the north service building (NSB) and the plant computer room. This links the simulator to the EOF/TSC using existing modems and equipment at the EOF.

Summary of Safety Evaluation: An automatic switch is installed in the computer room which allows either simulated or real data to be sent to the EOF and TSC. The automatic switch has a master key which enables/disables the ability to switch. Most of the time, the switch is in the disabled position with the key in the control room and real data is transmitted to the EOF/TSC. In the event of a power failure, the switch fails to the last switched position which is real data. The switch will be in the simulated position for emergency plan drills only. In the rare event of a power failure to the switch and a real plant emergency during an emergency plan drill, it would require manually overriding the switch to send real data to the EOF/TSC. This could be done in less than 10 minutes. NUREG-0696 states that the EOF/TSC availability must be made available within 30 minutes at all times. The NUREG also states a reliability of 0.01. The addition of the automatic switch does not impact the reliability.

A new cabinet (C195) is installed on the south wall of the computer room between the two rows of computer cabinets. The cabinet houses the equipment required for the data link to the NSB. The cabinet is installed to meet seismic Class 2/1 design criteria.

All cabling installed is plenum rated so as to not add a fire load to the computer room. Cabling is run either in cable trays below the floor or through seismic conduit above the ceiling. (SER 91-106)

3. MR 87-034*G, (Common), 480 V Motor Control Centers (MCCs). MR 87-034 installs amptector solid-state overload retrofit kits on 480 V DB breakers on load centers B01 through B04. MR 87-034*G includes retrofit to supply breakers for various MCCs including safeguards MCCs B32/B42.

Summary of Safety Evaluation: As breakers are retrofitted, the amptector settings are chosen so the time/current trip curve overlaps the original trip curve. Amptector setpoints for short delay time range from a minimum of 0.18 seconds to a maximum of 0.5 seconds. However, the short delay time setting for MCC supply breakers were 0.1 seconds. This requires that the new short delay time setting be raised to 0.18 seconds.

At a short delay time setting of 0.18 seconds, maximum fault current duration is increased. Calculation P-90-029 was performed to assure that cable damage curves for MCC supply cables lie above the MCC supply breaker trip curve. (SER 88-137-02)

4. MR 87-192*B, (Unit 1), Rod Position Indication System. The MR installs a digital rod drop and stepping test system. This is accomplished by permanently connecting a high-speed multiplexer to the test points of the Unit 1 control rod drive stationary, lift, and moveable coil power circuits and the RPI coil circuits.

Summary of Safety Evaluation: The modification replaces the Visicoder but does not disallow its use. The rod drop and stepping test system is comprised of two high-speed multiplexers; one MUX permanently connects to the back of the same test points in the rod drive circuitry that is used for rod stepping testing, and the other permanently connects 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) are high enough that the existing circuity does not effect the connection of the MUXs. The multiplexed signals are transmitted to a personal computer which generates plots of the signals from which a rod drop time can be determined. The system also allows verification of proper rod stepping and rod full out verification.

Since the new rod drop and stepping test system could affect no other systems except the existing RPI and rod control circuits, and since the only accidents and/or malfunctions have already been analyzed, it is concluded that the margin of safety is not diminished. (SER 91-069-02)

5. MR 87-193, (Unit 2), Rod Position Indication (RPI). The modification 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: The system replaces the Visicoder but does not disallow its use. It is comprised of two 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 is 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) will be high enough so the existing circuitry will not be effected by the connection of these MUXs. The multiplexed signals will be trasmitted to a personal computer which will generate plots of the signals from which rod drop time can be determined. The system will also allow verification of proper rod stepping and rod full out verification.

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 auctioneering 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.

Since the new system could effect no other systems except the existing RPI and rod control circuits, and since the only accidents and/or malfunctions have already been analyzed, it is concluded that the consequences of an any accident or malfunction of equipment important to safety is not increased. No margins of safety are effected by the proposed modification.

Therefore, the proposed modification does not involve an unreviewed safety question. (SER 91-069)

6. MR 88-076, (Common), Motor-Operated Valves (MOVs). The modification replaces the existing Limitorque SMB-000 motor operators with larger SMB-00 motor operators of the four motor-operated discharge isolation valves for the motor-driven auxiliary feedwater pumps (AFPs), P38A&B.

Summary of Safety Evaluation: The changes improve valve operability and the accuracy of the closed position indication.

The increased thrust provided by the larger SMB-00 motor operators provide additional margin to ensure that the associated valve shuts under maximum flow or opens under maximum differential pressure conditions. The increased thrust of the larger motor operators does not have a detrimental effect on the existing valves. The four-rotor limit switches enhance the accuracy of the valve position indication in the control room and improves valve operability by providing independent limit switch rotors for two functions previously wired off of a single rotor: closed valve position indication and open torque switch bypass. Since each rotor corresponds to one valve stem position, separating these two functions onto independent rotors allows for the closed position indication to correspond to actual valve seating, and allows the open torque switch to be bypassed longer, thus reducing the possibility of the motor tripping on open torque. The thermal overload circuitry for AF-4020-O, AF-4021-O, and AF-4023-O is modified to provide control room indication of an overload condition for these valve operators. The modification to the thermal overload circuitry has already been performed on AF-4022-O. The addition of T-drains allows for proper drainage of foreign liquids, equalization of pressure, and prevention of direct liquid or vapor intrusion into the motor operator.

The modification does not reduce a margin of safety during installation because both steam-driven AFPs and the opposite train motor-driven AFP are available to supply AFW to the steam generators, if required. Once installed, the margin of safety is increased because the increased thrust available from the larger operators provide more assurance that each valve opens or closes under the maximum differential pressure or maximum flow, respectively. (SER 88-129-01)

7. MR 88-188*G, (Unit 1), Motor-Operated Valves (MOVs). The modification installs 4-rotor limit switches and thermal overload indication on various MOVs. MR 88-188*G replaces existing 2-rotor limit switches with 4-rotor limit switches for the following MOVs in Unit 1: OS-001, OS-002, CV-313, CV-350, RC-427, RH-700, RH-701, RH-720, CC-815, CV-112C, CV-1299, MS-2019, CV-2189, CV-2190, SW-2880, CC-759A, CC-759A, CC-759B, SI-870A, and SI-870B. In addition, the modification changes the valve-position indication wiring to provide detection of motor thermal overload actuation for Unit 1 valves SW-2816 and SW-2817.

Summary of Safety Evaluation: The changes enhance the control room valve position indications for various safety-related MOVs and improves valve operability. The modification includes two types of MOV enhancements: replacement of existing 2-rotor limit switches with 4-rotor limit switches for 19 MOVs, and rewiring of the position-indication wiring at the motor control center to provide indication of the actuation of the valve motor's thermal overload devices for two valves.

The limit switch assemblies are procured as environmental qualification (EQ) equipment for EQ valve operators and EQ wire is used for installation on EQ valves. The additional rotors add negligible weight to the operator, and therefore does not affect the seismic analysis.

The modifications are installed during a shutdown refueling outage and/or in accordance with Technical Specifications. Once installed, the MOV enhancements decrease the consequences of an accident since control room operators have improved indication of valve position and improved valve operability. The MR also reduces the probability of equipment malfunction (i.e. an MOV not opening when required) because the MOVs open torque switch are bypassed longer. (SER 92-066-04)

8. MR 88-188*I, (Unit 2), Motor-Operated Valves (MOVs). The modification replaces existing 2-rotor limit switches with 4-rotor limit switches for the following MOVs in Unit 2: AF-4006, CC-0719, CC-0754A, CC-759A, CC-759B, CC-0815, CS-2189, CS-2190, CV-0313, CV-0350, OS-0001, OS-0002, SI-860C, SI-870B, SI-896B, SW-2880, SW-2908 and SW-2817.

In addition, the modification changes the valve position indication wiring to provide detection of motor thermal overload relay actuation for Unit 2 valves OS-0001 and OS-0002.

Summary of Safety Evaluation: The changes enhance the control room valve position indications for various safety-related and non-safety-related MOVs and improves valve operability. The modification includes two types of MOV enhancements: replacement of existing 2-rotor limit switches with 4-rotor limit switches for 18 MOVs and rewiring of the position indication wiring in the valve limit switch housings to provide actuation indication of the valve motor's thermal overload devices for the two MOVs.

The new limit switch assemblies and intermediate gear cases are procured as environmental qualification (EQ) equipment for EQ valve operators and EQ wire is used for installation on EQ valves. The additional components (rotors, gear cases) add negligible weight to the operator, and therefore do not affect the seismic analysis.

Once installed, the MOV enhancements decrease the consequences of an accident since control room operators will have improved indication of valve position and improved valve operability. They also reduces the probability of equipment malfunction (i.e. an MOV not opening when required) because the MOVs open torque switch is bypassed longer.

No new system components are being installed; the modification enhances the existing component functions and does not introduce the possibility of a new accident. Since the modification is internal to the components evaluated for malfunction in the FSAR, a malfunction associated with the modification could at worst cause the valve to be inoperable, which is already evaluated in th FSAR. (SER 92-066-05)

Summary of Safety Evaluation: This is a revision to SER 92-066-05, but does not completely supersede that document. The changes affect only work associated with MOV SW-2817. For other work associated with MR 88-188*I, reference SER 92-066-05.

Leaving SW-2817 in the open position (unable to be automatically or manually closed) while the 4-rotor modification and operator rebuilding work is performed does not increase the consequences of a previously evaluated accident or equipment malfunction. SW-2817 is Train B powered and receives a signal to close on a safety injection (SI) hereby isolating the non-essential water treatment service water load. The work on SW-2817 prevents this automatic action.

The accident which places the largest demand on the service water system is a loss-of-coolant accident (LOCA) resulting in a SI on one unit with the other unit in hot shutdown (or normal operation) with a loss of offsite power and a failure of G02 EDG. This accident does not take credit for closure of SW-2817 because of the loss of Train B power. The second most demanding accident scenario is the same SI but with failure of G01 EDG instead of G02. This scenario has the closure of SW-2817.

Additionally, an operator will be available to shut manual valve SW-527 (the upstream isolation to SW-2817) during the work; SW-527 verified as being capable of manual closure prior to the working SW-2817; both diesel generators are operable during the work; and all six SW pumps are operable during the work. A lake temperature of 60°F enables the containment coolers to perform their designed function with about 100 gpm less SW flow. It is very unlikely that the lake temperature will exceed 60°F at this time (late October/early November). The analysis for service water flow requirements during a LOCA does not include a break of non-seismic piping during a seismic event. As a prudent measure, SW-527 is throttled down to allow only normal water treatment demand. The work is expected to be completed and the system returned to normal operation within 48 hours. (SER 92-066-06)

9. MR 89-025*A, (Unit 1). Feedwater System. MR 89-025*A replaces the existing fixed orifice in the fourth pass drain lines from the MSRs with throttle valves. This allows the reheat steam flow through each MSR to be more efficiently controlled.

   Summary of Safety Evaluation: The application of these design inputs assures that the potential for a failure due to this change is the same as or less than the existing system. A failure of this line would create conditions similar to a steam line break and/or a loss of feedwater enthalpy accident. These accident analyses however, assume failures of a much larger magnitude and therefore, this change is not outside of the existing safety analysis.

   The high energy line break analysis addresses steam line failures in the general area of the change. This analysis assumes failures of a larger magnitude than possible with the change, and was determined that safe-shutdown capabilities are not affected.

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

10. MR 89-116 (Common), Emergency Diesel Generator (EDG). MR 89-116 adds ground detectors with associated local alarm lights and wiring to G01 and G02 including adding the alarm to control room common G01 and G02 trouble annunciator circuits.

   Summary of Safety Evaluation: The modification provides local ground alarms and control room annunciation for a ground on G01 or G02 when providing power to 4160 V switchgear independent of offsite power. The modification provides alarm and annunciation functions and is non-safety-related.

   All components added within the diesel generator control panels were seismically analyzed and seismically mounted to assure no seismic interaction hazards within the panels are created.

   The ground detection circuit is connected to the diesel generator output through fuses qualified as Class 1E isolation devices in accordance with IEEE-384. This assures that faults within the ground detector circuitry are isolated from and do not affect the ability of the diesel generator to perform its safety function. All wire and cable is fire rated in accordance with IEEE-383. (SER 93-001)

11. MR 90-006, (Unit 2), Main Control Boards. The modification removes the pressurizer low level channel alert circuitry. The SI portion of the circuitry was disconnected previously and was abandoned in place. The remainder of the circuitry still provides unnecessary annunciator and status light indications.

Summary of Safety Evaluation: The removal of the unnecessary alarms and the abandoned SI signal wiring does not cause safety concerns since these components are not necessary for plant safety since after TMI. The continuity of all wires installed to complete the daisy chain of safeguards control power is verified to ensure that no piece of equipment is left without power. The ORT 3 and 6 tests, along with the safeguards logic test, also verify that the safeguards control power is not broken. Therefore, the probability of an accident or malfunction of equipment important to safety is not increased either during or after the installation of the modification.

One train of the Unit 2 safeguards control power is electrically isolated during the removal of components and wiring for the pressurizer low level channel alert circuitry. All circuits that use the safeguards automatic sequencing of loads onto the buses still have manual operation available. Only one train of Unit 2 safeguards control power is out of service at a time. The work is performed during cold shutdown so the safety injection (SI) timing application for safeguards is not necessary. SI signals are normally blocked during cold shutdown. Service water availability is the limiting case during installation. The normal service water (SW) pump start on Unit 2 undervoltage is disabled for the Unit 2 train being worked on because the Unit 2 undervoltage sequencing uses the Unit 2 applicable train SI time delay relay which is out of service. Therefore, Unit 1 SW pumps A, B, and C are running to preclude any affects on normal SW operation due to a loss of normal power to the Unit 2 bus whose SW pump time delay relays are being worked on. The affected SW pumps still start on a Unit 1 undervoltage signal or a Unit 1 SI signal; therefore, SW system operability for Unit 1 is not impaired. The other shared safeguards load P38B (auxiliary feedwater pump) also starts on a Unit 1 SI signal. Thus, the consequences of an accident or malfunction of equipment is not increased.

When either train of safeguards control power is out of service, any containment penetration which cannot be isolated by the other train of safeguards alone is noted on Checklist 1E, Attachment A. The ability to isolate the containment before the reactor coolant system "time to boil" is maintained during the modification. The margin of safety for SW operability per Technical Specification 15.3.3.D is not degraded. (SER 93-068)

12. MR 90-086*B, (Unit 1), Containment. MR 90-086*B installs approximately 52' of handrails and one new ladder on the Unit 1 "B" steam generator shield walls. In addition, two existing ladders are revised to meet current OSHA standards.

Summary of Safety Evaluation: The structures are 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 clearances have been included in the design. State-certified structural welders preform the fabrication. Paints specified are compatible with the containment post-DBA atmosphere. The design conforms with OSHA specifications, as required by the FSAR and federal law.

The installation work plan does not permit activities when there is potential for introduction of debris into the reactor coolant system, such as when the reactor vessel head is removed and the debris screen is not installed. This restriction applies only to installation activities on the north exterior side of the "B" steam generator shield wall, as is the only location where installation activities occur that is also in close proximity to the refueling cavity. (SER 91-078-02)

13. MR 90-091, (Common), HVAC. This safety evaluation revises SER 92-048 that stated the containment openings in the removable panels would be smaller than the fans so the fans could not drop into the cabinets. Contrary to this, the fans were mounted on the bottom of the panels.

Summary of Safety Evaluation: MR 90-091 addresses high temperatures inside the plant computer multiplexing cabinets. With the cabinet doors closed, the cold junction box temperatures in the cabinets are continuously in high alarm. The cabinet doors must be left open in order to keep temperatures in the cabinets below the alarm setpoint. The MR also indicates that the room temperatures are too warm and installs forced ventilation (exhaust) fans in the removable panels at the top of cabinets C176/C177/C178/C179. In addition, air flow to the control room and computer room is rebalanced to improve cooling in the area of the multiplexing cabinets.

The additional weight of the fans (8 pounds each) does not adversely impact the seismic integrity of the cabinets or removable panels. Four machine screws, nuts, and lock washers provide the support so the fans do not drop into the cabinets during a seismic event. The final design was changed and a simplified calculation provided to document this. (SER 92-048-01)

14. MR 90-134*I, (Common), 125 Vdc System. The modification replaces the existing 125 Vdc switchboard D02, tie-in of new 125 Vdc swing buses (D301 and D302), battery recharging breakers (D110 And D111) and new class 1E swing battery (D305). The modification occurs after the new swing safety-related switchboard (D301) and battery have been successfully installed and tested (MR 90-134*E). The D301 switchboard is installed in the new switchboard/charger room.

Summary of Safety Evaluation: The new equipment was procured to meet or exceed the requirements of existing plant equipment. The new equipment was seismically qualified to function before, during, and after a seismic event and is installed in a seismically qualified structure. Separation is maintained between redundant trains during all phases of the installation. All Class 1E equipment for the modification is installed per the requirements of Specification PB-220.

The worst event that could occur during the live bus transfer part of the modification, (when the distribution panels D13 and D14 are connected to both buses D301 and D02) would be a loss of one DC train. This would result in an operating reactor trip and the loss of one train of safeguards control power. Administrative controls will be applied to ensure no work is being performed on necessary redundant train safeguards equipment at either unit during conditions requiring work on energized DC equipment. The loss of one train of safeguards control power does not increase the analyzed consequences of an accident or malfunction of equipment important to safety as evaluated in the FSAR. (SER 92-004-05)

15. MR 90-159*A (Unit 1), Feedwater. MR 90-159*A adds 17 drain/vent valves and two check valve bypass lines to the Unit 1 secondary piping systems. Four of the valves are on the moisture separator reheater (MSR) hotwell dump lines, four on the MSR stilling manifold dump lines, one on the heater drain tank pumps discharge line, two at the main feedwater regulating valve bypass valves, two on the condenser steam dump headers, one on the main steam to heating steam line, two on the high-pressure feedwater heater dump lines, and one on the heater drain tank drain to the condenser. The two check valve bypasses are for the main feedwater pump discharge check valves, and are in response to INPO SER 92-02.

Summary of Safety Evaluation: Each drain/vent/bypass connection is 3/4" NPS, and is located as close as possible to the low point of each of the referenced lines. Each connection allows the piping section to be drained and depressurized and the energy to be safely relieved prior to maintenance efforts.

The vent/drain connections provide a method of relieving the high-energy fluid from these lines so that seventeen air-operated valves (AOVs) may be more safely maintained, thus minimizing the potential for personnel burn injuries. The AOVs receive the vent/drain connections have been identified in the engineering evaluation for MR 90-159 as "high-maintenance items."

The connections comply with all requirements of USAS B31.1-1967 and other original piping system design requirements. These installations are non-seismic, non-QA, and non-safety-related. (SER 93-019)

16. MR 90-160*B, (Unit 2), Feedwater. MR 90-160*B adds six drain/vent valves and two check valve bypasses lines to the Unit 2 secondary piping systems. Two of the valves are on the moisture separator reheater (MSR) stilling manifold dr p lines, one is on the heater drain tank pump discharge line, two are on the high-pressure feedwater heater dump lines, and one is on the heater drain tank drain to the condenser. The two check valve bypasses are for the main feedwater pump discharge check valves, and are in response to INPO SER 92-02.

Summary of Safety Evaluation: Each drain/vent/bypass connection is 3/4" NPS, and is located as close as possible to the low point of each of the referenced lines. Each connection allows the piping section to be drained and depressurized and the energy to be safely relieved prior to maintenance efforts.

The addition of these connections do not affect any of the conclusions reached in the FSAR, the Technical Specifications, or any prior technical commitment to the NRC. The new connections comply with all of the requirements of USAS B31.1-1967 and other original piping system design requirements. These installations are all non-seismic, non-QA, and non-safety-related. (SER 92-073-01)

17. MRs 90-198, (Common), 90-198*A (Unit 1), 90-198*B (Unit 2), Boric Acid Heat Tracing. MR 90-198 replaces the boric acid transfer pumps heat tracing. The boric acid transfer pumps had heat trace cable sandwiched between insulation that leads to a long turnaround time for circuit repair and reinstallation. The modification installs heater strips that are easier to replace and provides enough heat to maintain the boric acid soluble (>145°F). Four heater strips, rated at 150 watts each are connected to each circuit (primary and secondary).

Summary of Safety Evaluation: Based on calculations, the supply breakers do not overload the new heat trace installed. The modification does not affect emergency diesel generator loading as the diesel generator loading calculation is based on the boric acid heat trace supply transformer rating. MR 90-198 is worked in conjunction with MR 91-133 which installs removable pump enclosures to allow manual vibration monitoring. The plant conditions for installation of MR 90-198 are governed by MR 91-133. Following installation, the heat trace circuits are checked for shorts and grounds. Acceptance testing consists of a functional test of the circuits and a check of the thermocouple. (SER 92-099)

18. MR 90-259 (Common), Chemical and Volume Control System (CVCS). The modification installs lockable hinged covers over the skirt manway on the CVCS holdup tanks. This covers barricade access to the skirt region under T8. The installation of these barricades allows the deposting of a large portion of the CVCS holdup tank cubicle.

Summary of Safety Evaluation: MR 90-259 blocks access to the skirt region of the CVCS holdup tanks. The design is simplistic and accomplishes the intent with minimal impact on the tank and support skirt. This is accomplished by welding a barrier assembly to the manway coaming. It does not affect the tank itself nor its pressure boundary. The modification adds an insignificant 20# to the skirt of the tanks.

This modification limits access to the skirt region; however, there is no reason to access under the tank because no equipment is located under the tank. (SER 92-113)

19. MRs 91-011, (Unit 1) 91-012, (Unit 2), Component Cooling (CC) Water. MRs 91-011/012 install a manual handwheel on the air operator of throttle valves 1&2TCV-130, to limit the flow to the maximum design value.

Summary of Safety Evaluation: The handwheel is set to restrict the maximum flow through this line to its maximum design flow on the loss of IA. The restriction reduces the flow demand on the CC pump during containment sump recirculation operation to prevent exceeding the design flow of the pump with only one train available. The handwheel was purchased from the valve operator's manufacturer and is compatible with the valve and operator.

The inlet and outlet valves are used to isolate the 1&2TCV-130 throttle valves during the installation of the handwheel. This secures cooling flow to the non-regenerative HXs. The installation occurs during an outage, with the unit in refueling shutdown. Component cooling to the non-regenerative HXs is not required at this time. The residual heat removal HXs maintain the letdown flow temperature below the 145°F maximum for the demineralizers.

The modification does not create any new accident scenarios or failure modes, different from the original design, during installation or operation. (SER 93-020)

20. MR 91-057 (Unit 1), Component Cooling (CC) Water. MR 91-057 upgrades the existing CC water piping supports to meet seismic analysis requirements. The modification is performed so the current system capacity is not degraded nor the systems functionality impaired.

Following is a description of individual piping support modifications:

ACC2C3002R: The gang support required the addition of two flat plates to be added to the existing trapezes to stiffen the piping support and prevent dynamic uplift on the upper pipe.

ACC2C3001G, ACC2C3003G, ACC2C3005G: These piping supports require replacement of the piping standard U-bolt with a heavy-duty U-bolt.

AC-152N-1-H120: The piping anchor required grouting of the floor penetration.

AC-152N-1-S-476: The piping support required an increased fillet weld bead at four locations on the support and grouting of the floor penetration.

AC-152N-1-S478: The support required the replacement of structural members welded to the pipe pressure boundary and grouting of the floor penetration. The welding to the pressure boundary is within the currently accepted practices for welding defined in the WPS-1 procedure and shall not impact the operability of the system during modification.

Summary of Safety Evaluation: MR 91-057 upgrades CC water pipe supports to meet qualification requirements for seismic Category 1 piping supports. The piping systems and all associated piping supports meet the applicable stress allowables.

Visual non-destructive examination (NDE) was required for the welded connections. NDE shall be performed by the contractor under the guidance of the QAS and ISI groups. Documentation of the acceptability of the welds was made at the completion of the support modification and documented on the traveler. (SER 92-108)

21. MR 91-085 (Unit 1), Computer. MR 91-085 installs necessary wiring to make intermediate range startup rates available to the computer. This involves disabling the rate and comparator drawer in 1C130 during installation.

Summary of Safety Evaluation: The modification addresses a deficiency in the safety assessment system critical safety function subcriticality status tree. The modification allows a backup physical hardware input to the critical safety function status tree ST-1 allowing the logic to be based on the exact hardware that the operators use when N40 fails. This involves tapping off existing signals in the rate and comparator drawer of NIS cabinet 1C130 and terminating existing wiring from the computer multiplexers.

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

Performing the modification during the Unit 1 outage when there is no fuel movement puts the unit in a state where these indications are not required and the deviation alarm can be ignored. (SER 92-097-01)

22. MRs 91-109 (Unit 1), MR 91-110 (Unit 2), Component Cooling (CC) Water. The inservice test (IST) program requires both suction and differential pressure measurements to be taken quarterly for CC pumps P11A&B. The modifications install a permanent suction pressure gauge for each of the four CC pumps. Permanent pump suction pressure gauges provide continued satisfaction of IST program requirements and improves the ease and efficiency of the quarterly test.

Summary of Safety Evaluation: The gauges are normally isolated from system pressure, and are only used during quarterly testing. This minimizes personnel exposure to chromated water during quarterly testing.

The CC system outside of containment is a closed system for containment integrity purposes. The permanent suction pressure gauges are located downstream of existing pump casing drain valves, CC-790A&B that are normally shut and provide for system integrity. The permanent suction gauges do not adversely affect system, and consequently containment integrity, nor does it adversely affect the operability of the CC system. The design consists of mechanical joints including stainless steel tubing and swagelok fittings. The resulting configuration is seismic. The temperature and pressure rating of the new components meet or exceed original system design criteria. The modification does not present an unreviewed safety problem as the design function of the CC system is not impaired. (SER 92-102)

23. MR 91-113 (Common), Fuel Handling System. MR 91-113 disables the indication functions associated with the engage and disengage limit switches (LS2 and LS3) on the spent fuel pool (SFP) control rod handling tool (Z23H) and removes the indication light fixtures from the tool's control box.

Summary of Safety Evaluation: The SFP control rod handling tool (Z23H) was initially designed with underwater limit switches to indicate engagement/disengagement of the tool to the rod cluster control assembly (control rods). These limit switches (LS2 and LS3) have repeatedly failed due to water environment of the SFP. The modification allows for the reconfiguration of the SFP control rod handling tool's control box wiring and indication. The two limit switches that have repeatedly failed are disabled and their respective indication wiring and lights removed from the tool's control box. The functions of the remaining limit switches associated with the tool are not affected by this modification. The operability and function of the tool for fuel handling is not hampered or rendered less effective by this change. Procedure OI-75, for control rod handling, utilizes a better method of verification of the tool's engagement on a control rod by using the SFP bridge load cell. This method provides a positive indication of the tool's engagement/disengagement on a control rod due to the increase or decrease of weight indicated on the load cell. The limit switches only indicate the tool's gripper position, not actual control rod engagement. (SER 92-032)

24. MR 91-116*B, (Common), Main Control Boards. MR 91-116*B relocates 13.8 kV system control (G05 combustion turbine and all 13.8 kV breakers except H52-06, -30 and -32 which are associated with Unit 2) and Unit 1 345 kV system control (Breakers 1F89-112 and 1F89-122B) to the rear of C02 to provide space on the front of C02 for the future addition of the new emergency diesel generators' control. Additionally, 345 kV switchyard indication for 1F52-122, F52-111, F52-Q303, F52-121, F52-123, F52-BS-1-2 and F52-BS-2-3 are rearranged on the front of C02.

Summary of Safety Evaluation: MR 91-116*B does not change the operation or failure modes of the relocated 13.8 kV and 345 kV system controls and these controls do not directly affect components or equipment required for controlling post-accident doses. The controls relocated by this modification do not perform a safety-related function but are located in a panel with safety-related components.

A seismic analysis was performed to verify the control panel seismic qualification and concluded no reduction and no seismic interaction hazards were created. The component mountings were seismically qualified using the seismic qualification utilities group (SQUG) methodology. During installation, cutting and drilling of C02 front and rear occurs which results in some panel vibration. Based on experience of past control board modifications, this is not expected to result in the malfunction of any control board equipment.

Operational aids are installed for the 13.8 kV and 345 kV control and indicating devices associated with Unit 2 to help prevent operator error. If these new switches are manipulated, no system misalignments or equipment operations occur since these controls are not connected. (SER 93-025)

Summary of Safety Evaluation: This is a revision to SER 93-025 and addresses portions of the work with Unit 1 at power and the possible use of a new control board cutting tool.

A new cutting device may be used which has been used at other nuclear power plants may be used. Based on their past control board modification experience using this device and on our own control board modification experience, the vibration is not expected to cause a malfunction of any control board equipment. The new cutting device is a hydraulically-driven milling machine where the compressor is located outside the control room to reduce the noise levels. Hydraulic lines run from the compressor to the location of the cuts where the milling device is mounted on the control board. The mounting of the milling device is analyzed to verify that no seismic interaction hazards are created. The likelihood of a hydraulic line rupturing is very small. In the event that a line did break, no plant equipment is expected to fail or malfunction. Splattering of hydraulic fluid due to a line breaker could have chronic impacts on plant equipment made of certain plastics. Contact blocks on control switches may be affected and require replacement. Precautions are taken to minimize the likelihood of a hydraulic line rupture and to minimize the impact if one were to rupture. Precautions are also taken to contain any stray metal chips created by the cutting to prevent them from entering into other plant equipment and creating electrical fault paths. (SER 93-025-01)

25. MR 91-116*C, (Common), 13.8 kV and 345 kV System Control. The systems affected by the modification are the 13.8 kV distribution system, and the 345 kV switchyard. These systems provide sources at offsite (and auxiliary power) and connect the main generators to the grid and are associated with the accident analysis for loss of all AC and loss of external load.

Summary of Safety Evaluation: The final design configuration for this system is identical to the existing configuration with the exception that Unit 2 13.8 kV and 345 kV controls originally on the front of C02 are now located on the rear of C02. Additionally, G04 subpanel was installed on C02 front to accommodate a future addition of a new diesel generator. Therefore, the final design results in no impact on the existing safety analysis, Technical Specifications or margin of safety.

During installation and testing, interim system configurations occur. Specifically, breaker and disconnect switch control for the relocated control switches are placed in "Local" during which time breaker and disconnect switch position indication are only available locally. All of these conditions have been analyzed to confirm that there will be no impact on existing safety analysis, Technical Specification or margin of safety.

Components (and their mounting) added within the panels were seismically analyzed utilizing the SQUG methodology and will be mounted to assure that no seismic interaction hazards within the panels are created in the final design. During installation, failures could potentially result in loss of offsite power and loss of external main generator loading. Both of these conditions were analyzed as part of the existing safety analysis.

The C02 front panel was seismically analyzed to verify seismic qualification of the panel with the added components is not reduced (Calculation 6704.001-C-064). (SER 93-025-03)

26. MR 91-133*B (Unit 1), Chemical Volume and Control System (CVCS). MR 91-133*B installs monitoring instrumentation necessary to trend boric acid transfer pump performance as required by the Inservice Testing Program. The instrumentation includes a magnetic flow tube and transmitter, a suction side pressure gauge, modification of the old boric acid filter inlet pressure gauge to allow calibration for use as a pump discharge pressure gauge, and a removable pump casing that allows manual vibration monitoring.

Summary of Safety Evaluation: The components have no affect on the operation of the boric acid system. The flowtube is installed just upstream of the tee separating the normal and emergency flow paths. This allows the flow transmitter to be used as a local indication of flow whenever desired by the control room. All areas that contain concentrated boric acid are heat traced to ensure the boric acid remains soluble (>145°F). The installation occurs during the Unit 1 refueling shutdown. The interim conditions during the shutdown isolates the system to allow it to be flushed and drained, and the heat tracing deenergized. In this condition, it is not possible to add concentrated boric acid using the normal boration path. A boration flow path from a boric acid storage tank or refueling water storage tank to the reactor coolant system through the safety injection pumps exists at all times in accordance with TS requirements. The work scope was reviewed as a part of the shutdown risk assessment. (SER 92-093-01)

27. MRs 91-178 (Unit 1) and 91-179 (Unit 2), Gas Stripper. The modifications install a conversion kit on the Unit 1 and 2 letdown gas stripper prefilters, 1F60A and 2F60B, to facilitate the replacement of the existing six filter elements with a single filter cartridge.

Summary of Safety Evaluation: The original filter elements used in these filters were of a tubular wound, synthetic fiber design and a 5 μm pore size. The replacement filter cartridge is an epoxy-coated, glass fiber, pleated sheet design with a 2 μm pore size. The exopy-coated glass fiber filter media is more effective at removing smaller particles and is more durable than the original media.

Replacement of the current filter elements with a filter cartridge with a smaller pore size does not change the mode of failure. The loading of the filter will however, occur at a faster rate. Over operating history, the 2F60B letdown gas stripper prefilter was changed three times whereas, no record of an 1F60A change was found. The low load rate of these filters therefore, indicates the existing filter media is relatively ineffective. The use of smaller pore size filter media therefore improves the ability of the filter to effectively remove smaller particulate contamination and reduces contamination within the reactor coolant system.

FSAR Table 11.1-4 states the retention size of the letdown gas stripper prefilters is 25 μm. The actual filter media used by these filters has always been 5 μm. Thus, although this modification results in a deviation from and a change to what is listed by the FSAR, the retention size of the letdown gas stripper prefilters has always been less than that listed. Future reductions in the pore size of the filter media requires FSAR changes. The retention size listed in the FSAR Table 11.1-4 will therefore be changed to $\leq$ 5 μm to avoid future changes to the FSAR. (SER 93-006)

28. MRs 91-199 (Common), 91-199*A, (Unit 1), 91-199*B, (Unit 2), Auxiliary Safety Instrumentation Panel (ASIP). MR 91-199 provides an ASIP alarm on high priority PPCS alarms by installing the hardware needed to interface the plant process computer system (PPCS) output relay chassis to the respective 1C20/2C20 ASIPs alarm windows. Each unit has its own alarm window on its respective ASIP.

Summary of Safety Evaluation: MR 91-199 adds two wires in each ASIP to connect each PPCS output chassis to the desired alarm window on each unit's ASIP. Programmed PPCS alarms annunciate the ASIP alarm, alerting operators to high priority PPCS alarms. The margin of safety of the units is improved since operators become aware of off-normal plant conditions or potential Technical Specification threatening conditions in a more timely manner.

The annunciator circuit is actuated from an existing PPCS output relay which operates via PPCS software. Failures of the annunciator circuit do not affect PPCS operation. Alarm circuit wiring is non-safety-related, non-train-related, and is routed in non-safety-related wire bundles within the ASIP. All wiring is inside the ASIPs, so no Appendix R considerations exist.

The seismic qualifications of the 2C20 panel is not affected since no modifications are required to the 2C20 structure. (SER 92-107)

29. MR 91-209*A (Unit 1), Main Steam. MR 91-209*A installs new valve components on the main steam isolation valves (MSIVs). This reduces the amount of force required to fully shut the valve, and increases the forces available to fully shut the valve.

Summary of Safety Evaluation: The uplift force provided by the new 12" cylinders and stronger spring are greater than that provided by the old 9" cylinders. The resulting force is more than the originally supplied 9" operator. It is not possible to quantify the exact uplift force required without performing specific force testing at 100% power. The control and function of the valves are not impacted. The valve manufacturer performed the design that includes potential failure mechanisms. The new design does not impact the ability of the valve to close tight under steam pressure as required by the tube rupture accident analysis. It improves the ability of the valve to shut within 5 seconds as required by the steam pipe rupture accident analysis and Technical Specifications.

The preload is increased from 770 lbf to 2500 lbf, with the full load increased from 1700 lbf to 3900 lbf. The new configuration provides greater assurance that the valves will fully shut upon demand. This design halves the packing friction by removing it from the non-operator end. The ball thrust bearing is required because the shaft does not pass through the bearing housing, and therefore sees steam pressure at its end. The ball thrust carries this shaft load. The valves are tested by performing preacceptance testing to check the valve assembly, and the inservice tests for these valves.

The modification is performed during cold shutdown <200°F conditions. In addition, the steam generator secondary side manways and handholes must be installed when containment integrity is required. Under these required conditions, the modification has no impact on the operation or safety of the plant. (SER 93-024)

30. MR 91-221 (Common), Computer. MR 91-221 revises the power fail alarm circuit for the containment hydrogen monitors such that the power fail alarm seals in until a local reset button manually clears the alarm. The modification applies to both the white and yellow hydrogen monitors. Although the control room annunciator alarms on monitor power loss, it can be cleared upon power restoration. The seal in circuit requires I&C or Operations to locally reset the alarm after calibration constants are reentered.

Summary of Safety Evaluation: The modification does not affect the hydrogen monitor interface to the plant process computer systems (PPCS) or to the C20 auxiliary safety instrumentation panel (ASIP) indication, and all original alarm inputs remain unaffected. Redundancy between the monitors is maintained, with no Appendix R concerns present. The modification concludes seismicity of the monitors, as well as that of the C174 and C175 cabinets, remains unaffected.

The modification requires that each hydrogen monitor be deenergized, one at a time, to allow wiring and testing of the alarm circuit change. Work controls minimize each monitor's downtime. Technical Specifications state that if one of the redundant monitors is inoperable, then a 30-day LCO is in effect. Technical Specifications also require that at least one detector channel be operable per unit. Note, that having one unit shutdown does not provide an advantage since each monitor supports two Unit 1 channels and two Unit 2 channels. Benefits received from monitor downtime outweighs any associated risk of having a monitor out of service. Work may be done on one monitor at a time, with both units at power without reducing the margin of safety defined in Technical Specification Bases. (SER 93-047)

31. MRs 92-004 (Unit 1), 92-005 (Unit 2), Safety Injection (SI). MRs 92-004/005 install a pressure gauge downstream of each SI pump discharge pressure transmitter. Each new pressure gauge is used to determine pump discharge pressure for SI pump testing required by Technical Specifications.

Summary of Safety Evaluation: Pressure gauges are installed during normal plant operation. Much of the installation is completed with no impact on plant operations or systems. The final connection between the new pressure tap and the transmitter pressure tap must, however, be installed with the pressure tap isolated from the SI system. While a transmitter is isolated, control room indication for the associated SI pump is out of service. Control room indication for SI pump discharge pressure is provided to permit control room monitoring of the effective operation of the SI pumps following an emergency core cooling system actuation signal. Thus, while a transmitter is isolated, another method must be used to verify SI pump operation (following any SI signal).

The modification does not affect the operation of the control room pressure indication for SI pump discharge pressure, and therefore does not affect the ability to monitor effective SI pump operation from the control room. The ability to mitigate an accident requiring emergency core cooling actuation is not impacted by this modification.

The pressure taps for the new gauges are installed by the same requirements of the original pressure taps and are tested prior to returning an isolated transmitter to service. Installation of this modification does not require new penetrations into the SI or other primary system. New tubing, fittings, and pressure gauges are also seismically secured to prevent damage to the surrounding pressure transmitters during a safe shutdown event. This modification does not introduce new mechanisms nor increases the probability for the malfunction of a system or equipment important to safety. (SER 92-104)

32. MR 92-025*C, (Common), Main Control Boards. The modification provides a simplified means for calibrating the D105 and D106 battery ammeters. The modification removes the requirement and makes calibrations faster and safer to perform by installing rotary ammeter calibration switches ("test switches") and 5-way binding posts ("test jacks") into each ammeter circuit.

Summary of Safety Evaluation: The test switches, test jacks, and terminal blocks are of inherently rugged construction, and are identical to components used elsewhere in the plant. Furthermore, these components were seismically qualified, and meet seismic Class 1E requirements by using QA-qualified hardware and materials. Proper cable separation and lugging requirements were observed. Therefore, the Class 1E qualification of the D03 and D04 panels are not affected.

The rugged construction and mounting ensures that the possibility of a failure is extremely remote. Furthermore, any such failure would not affect the ability of the D03 and D04 buses to supply their safety-related loads. An open- or short-circuit failure in one of the components result only in a loss of battery charging current indication on the associated ammeter. However, even with the ammeter out of service, battery discharge is still indicated by the receipt of a bus undervoltage alarm if bus loading exceeds battery charger supply capability. The only other mode of failure for these components is a short-circuit-to-ground fault. The D03 and D04 buses are ungrounded to prevent a single ground fault of this type from causing a loss of bus voltage. Furthermore, each bus features a ground fault detector (built into the online charger) which triggers an alarm if any such failure was to occur.

The configuration of the D03 and D04 buses is such that it is impossible to isolate the D105 and D106 ammeter current shunts without first deenergizing the associated DC buses, an action which is not permitted under Technical Specifications. The result is that the installation must take place while the associated D03 and D04 buses remain energized. However, to minimize personnel risks, the installation is structured such that no work takes place on or immediately near the energized current shunts or buswork. A worst-case scenario during installation involves personnel error resulting in the complete loss of only the D03 or D04 bus. The procedure contains controls to ensure that the probability of this occurring is extremely small. Furthermore, installation activities are permitted to proceed in only one train (D03 or D04) at a time, so even a worst-case event would be essentially equivalent to a single-failure fault on one of the DC buses, with the opposite train of DC power being unaffected. The loss of a single train of DC control power has already been accounted for in FSAR accident analyses. (SER 92-036-02)

33. MR 92-085*A, (Common), Emergency Diesel Generator (EDG). MR 92-085*A adds a new 22" diameter flexible connection to the existing G01 EDG exhaust piping (Line HB-29) at the turbocharger outlet flange connection. Two existing spring hangers are removed and replaced by two new pipe supports.

Summary of Safety Evaluation: The new flexible connection is compatible with the existing system and suitable for use in the EDG exhaust pressure and temperature. During installation, G01 is removed from service and requirements for a limiting condition of operation (LCO) with one diesel generator out of service per TS 15.3.7.B.1.g is maintained.

The modification does not alter the G01 operation or design basis when completed. It does not increase existing or create new accident scenarios or probabilities of equipment malfunction. (SER 93-053-01)

34. MR 92-085*B, (Common), Emergency Diesel Generator (EDG). MR 92-085*B adds a new 22" diameter flexible connection to the existing G02 EDG exhaust piping (Line HB-29).

Summary of Safety Evaluation: MR 92-085*B adds a new 22" diameter flexible connection to the existing G02 EDG exhaust piping (Line HB-29) at the turbocharger outlet flange connection. Two existing spring hangers are removed and replaced by two new pipe supports.

The new flexible connection is compatible with the existing system and suitable for use in the EDG exhaust pressure and temperature. During installation, G02 is removed from service and requirements for a limiting condition of operation (LCO) with one diesel generator out of service per Technical Specification 15.3.7.B.1.g. is maintained.

The modification does not alter the G02 operation or design basis when completed. It does not increase existing or create new accident scenarios or probabilities of equipment malfunction. (SER 93-053)

35. MR 92-091, (Unit 1), Auxiliary Feedwater. The steam-driven auxiliary feedwater pump (AFP), mini-recirc valve, AF-4002, was added to the 1990 Inservice Testing (IST) program. The program requires the fail-safe test and stroke test - close AF-4002 on an annual basis, during cold shutdown. MR 92-091 adds an air line which is routed around the AOV control solenoid valve, AF-4002-S, and provides instrument air (IA) to the AOV for the required testing.

Summary of Safety Evaluation: The two Whitey 2-way ball valves control the flow of IA to and from the AOV during testing (opening and closing the AOV). The ball valves and their positions are labeled and, during normal operations, the ball valves are locked in their required positions.

MR 92-091 does not increase the probability of IA failure to the AOV. Should the new installation fail, the consequences would be no different than those originally designed for; the valve would fail shut.

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

36. MR 92-093 (Common), Auxiliary Feedwater. Mini-recirculation valves, AF-4007 and AF-4014, for each electric auxiliary feedwater pump (AFP) were added to the 1990 Inservice Testing (IST) program. The program requires the fail-safe test, the full stroke - open (BT-O), and the full stroke test - close (BT-C) of AF-4007 and AF-4014. AF-4007 and AF-4014 are air-operated valves (AOVs) that fail shut upon loss of instrument air (IA). The modification provides the means to allow the required testing to be performed.

Summary of Safety Evaluation: The Whitey 2-way ball valves control the flow of IA to and from the AOVs during testing. The ball valves and their positions are labeled and, during normal operations, the ball valves are locked in their required positions.

The installation increases the probability of IA failure to the AOV, in that components have been added to the IA lines. Should the installation fail, the consequences would be no different than those originally designed for; the valves would fail shut. However, the design pressures and temperatures of the pressure retaining components used in this modification exceed the specifications of IA, and all new fittings are leak checked to assure leak tightness prior to return to service. This helps to minimize the potential for any leakage or subsequent failures.

During AOV operation, a failure would be indicated at C01 in the control room by the AOV shut indication. At other times, a failure would be detected locally by the auxiliary operator.

The installation is of short duration with the out-of-service time for each electric AFP held to a minimum, and each pump returned to service within the 7-day limiting condition for operation (LCO) allowed by Technical Specification 15.3.4. The installation does not change the normal operation of AF-4007 and AF-4014, while it allows the required testing to be performed. Its failure causes the AOVs to go to their originally designed for fail-safe position and is evident at the control boards or locally. There are no adverse system interactions. This work is scheduled while in LCOs for other work. (SER 93-022)

37. MRs 92-101 (Unit 1), and 92-102 (Unit 2), Service Water (SW). MRs 92-101/102 add the 1&2MS-2090-S valves to the Inservice Testing (IST) program. This requires local actuation and local position indication of the solenoid valves. These valves control the SW cooling to the bearings of the steam-driven auxiliary feedwater pumps (AFPs).

Summary of Safety Evaluation: The current solenoids are replaced with solenoid pilot, air-operated valves (AOVs). The new configuration has a signal box which mounts on the top of the AOV, with contacts that run off the shaft that make or break, depending on the shaft position. The signal is run to a test panel to provide local instrumentation. When the switch is at "normal operation," the solenoid pilot is energized and open, allowing instrument air (IA) to close the valve. If the switch is turned to the "test" position, the current is cut closing the solenoid pilot, and the valve is open. With a loss of IA or AC power, the valve goes to its fail-safe position, open.

The new valves installed perform the same function under the same conditions as the old valves. The valves are constructed of the same material. Per the previous design, the valves open on a sump start signal. The valves also fail-safe open with either a loss of IA, or a loss of AC power. All new electrical and service water installations are seismic. No work is performed on other AFPs during each modification. (SER 93-023)

38. MR 92-162 (Unit 1), Reactor Coolant System (RCS). MR 92-162 replaces the existing spring piping support, RC-2501R-Z-RC2, on the pressurizer surge line (PSL) with a spring piping support to allow greater thermal travel of this piping. The PSL is out of service during the installation due to the refueling outage. The PSL piping and supports meet all applicable pipe stress and support load Code requirements.

Summary of Safety Evaluation: The current PSL piping and supports were analyzed per Westinghouse WCAP-13509. While the analysis shows the current system is operable, modifying support RC2 to allow greater thermal travel increases the system's ability to withstand operating loads and stresses while performing its intended design function. The modification is to be performed during U1R20 when the PSL is out of service. Temporary support of the PSL piping is provided during the modification and controlled by the IWP.

The PSL piping and supports meet all applicable Code requirements. The modification does not involve an unreviewed safety question. (SER 93-014)

39. MR 92-163 (Unit 1), Containment Pipe Supports. MR 92-163 upgrades two existing pipe supports inside the Unit 1 containment structure to meet thermal and seismic analysis requirements. These design changes are required to meet commitments made as a result of NRC Bulletin 79-14.

Summary of Safety Evaluation: The current as-built piping and pipe supports for the 4SI-1501R-3 safety injection (SI) line were evaluated for their ability to withstand the design basis loads and stresses imposed on the system. The results show that while the system is operable as is, support SI-1501R-3-H6 requires an addition of two plates to reinforce its existing U-bolt and SI-1501R-3-S230 requires replacement of its existing U-bolt with an angle iron. The Code compliance analysis for the 4SI-1501R-3 piping is documented in WE Piping System Qualification Report No. 100096. All Code stress allowables for piping and supports are satisfied. Temporary support of the piping is not required, and the piping system remains operable throughout the installation process.

Both of these design changes enhance the 4SI-1501R-3 piping capability to withstand the design basis loads and are appropriate for the current conditions. (SER 93-013)

40. MR 93-026, (Unit 2), Main Steam. MR 93-026 modifies main steam pipe support EB-1-MS-2H7.

Summary of Safety Evaluation: During a walkdown of the main steam piping the pipe clamp nut on main steam piping support EB-1-MS-2H7 (2H7) was found in contact with the supporting channels, thus restraining upward movement, especially thermal movement, of the main steam piping. Operability of the main steam system is documented in Piping System Qualification Report 200082, Attachment G. MR 93-026 modifies the existing pipe clamp on main steam "A" train piping support 2H7 to allow greater vertical travel of the main steam piping while maintaining deadweight piping support. Support 2H7 is located inside Unit 2 containment, El 102'. Work was performed during U2R19. The modification includes temporary deadweight support of the main steam piping, removing the asbestos insulation, cutting the 3-bolt clamp to convert it to a 2-bolt clamp (to eliminate the interference), resetting the hot/cold loads of the spring can, and reinsulating the piping.

The modification brings the main steam piping and its supports within Code allowables for piping stress and support loading. It enhances the 2H7 spring hanger's ability to perform its intended design function of allowing free thermal travel of the main steam piping by removing the interference between the pipe clamp nut and the support channels, while providing deadweight support to the main steam piping.

No thermal movement is expected. No new accident initiators are introduced. The modification does not introduce any new failure modes to the support or piping than those already considered. (SER 93-061)

41. MR 93-028 (Unit 1), Chemical Volume and Control System (CVCS). MR 93-028 replaces the existing 2" check valve 1CV-369B and adds a 3/4" test connection in the residual heat removal (RHR), CVCS crossconnect line to test containment isolation valve 1CV-369A. 1CV-369B is replaced because of the potential for excessive leakage through the valve. There are no interim configurations or conditions.

Summary of Safety Evaluation: A 3/4" test connection and root valve 1CV-369C are added upstream of 1CV-369A to perform a seat leakage check on 1CV-369A. This ensures conformance to the existing design criteria for containment isolation and provides means for testing the seat leakage on 1CV-369A. (SER 93-039)

42. <u>MR 93-030 (Common)</u>, Service Water (SW). The 4" service water piping from the outlet of the component cooling heat exchanger TCVs to, and including, the 4" isolation valves are replaced with stainless steel piping (originally carbon steel) and a ball valve (originally a gate valve). This reduces the susceptibility to erosion damage.

<u>Summary of Safety Evaluation</u>: The final piping configuration is similar to the old piping, differing only in that the materials are stronger and the replacement valve is less susceptible to internal damage. During installation, a 4" pipe is opened while unisolated from the SW discharge header. This presents two real potential concerns. The first is that air may be sucked into the discharge header, which is normally at a vacuum. This air eventually makes its way to the condenser discharge water box, where it could cause air-binding of the condenser circulating water and a resultant loss of vacuum on the condenser. This is not a safety issue - the condensers are not relied upon for accident mitigation.

The second concern is the unlikely possibility of flooding from the unisolated 4" SW line. This is unlikely because the SW discharge header is normally at a vacuum. Even if flooding were to start, there will be workers on the scene with pipe plugging equipment ready to stop the flooding. In no case would there be enough water to impact the safety of the plant (43,000 gallons are necessary to endanger the RHR pumps - see NEPB 87-250, which evaluates the consequences of flooding in the PAB). <u>(SER 93-052)</u>

43. <u>MR 93-039, (Unit 2)</u>, Containment Spray. MR 93-039 modifies three existing containment spray pipe supports inside the Unit 2 containment structure to meet thermal and seismic analysis requirements. The support upgrades consist of eliminating support SI-301R-1-2S248, removing a strap from support SI-301R-1-267 and removing an uplift angle iron support from SI-301R-1-2H22.

<u>Summary of Safety Evaluation</u>: The current as-built piping and pipe supports for the 6"-I-301R-1 containment spray line were evaluated for their ability to withstand the design basis loads and stresses imposed on the system. The results of the analysis show that, while the system is operable as is, a loss of coolant accident or a main steam line break would impose thermal stresses on the piping in excess of Code allowable values. These stresses can be eliminated with the removal of support SI-301-R-2S267 and the modification of supports SI-301R-1-2S248 and 2H22. The Code compliance analysis for the 6"-SI-1501R-3 piping is documented in Piping System Qualification Report Nos. 200073 and 200074. All Code stress allowables for piping and supports will be satisfied. Temporary support of the piping is not required, and the piping system remains operable throughout the installation process.

The design changes enhance the 6"-SI-301R-1 piping's capability to withstand all thermal loads without compromising the piping's seismic integrity and are appropriate for the current conditions. <u>(SER 93-063)</u>

TEMPORARY MODIFICATION

1. <u>TM 93-012 (Common)</u>, Service Water (SW). TMs 92-007, 92-009 and 93-012 install blank flanges upstream of TCV-12A, TCV-12C and TCV-12D. These temperature control valves are on the SW outlet side of the component cooling (CC) water heat exchangers (HXs) in the bypass line around the 12" main outlet piping.

   <u>Summary of Safety Evaluation</u>: This evaluation is an addendum to SER 92-014 which evaluated TMs 92-007 and 92-009. TM 93-012 allows the SW outlet bypass line from the CC water heat exchanger, HX-12C, to be isolated by installing a blank orifice flange in the same manner as described in SER 92-014. The blank orifice flange is of similar dimensions and material as stated below. Removal of the blank orifice flange occurs upon completion of the replacement of the leaking piping section. The conclusion that this does not involve an unreviewed safety question is unchanged from the original evaluation.

   The structural integrity of the seismic SW piping is not degraded with the blank flange installed since the additional weight and change in mechanical joint strength is insignificant. It can also be noted that with the blank flange in place, the possibility of flooding, if the TCV failed open and the downstream piping was open or the piping ruptured, is eliminated.

   The piping in which the blanks are installed are Bechtel Line Class JB which specifies a 150# ASA ASTAM A-181 flange. A 150# stainless steel Type 304 blank orifice flange with flexitallic gaskets on either side is used. The use of stainless steel Type 304 flanges in the SW system was evaluated and found acceptable in MR 87-158. A non-QA flange is dedicated for use in the QA system since QA-scope flanges are not in stock. The blank flange is given a restricted QA release number.

   A stress calculation was performed to determine the effect of an unsupported TCV (if downstream pipe breaks) on the stub pipe connection to the 12" SW pipe. The calculation determined the valve did not have to be supported to ensure integrity of this pipe. Based on engineering judgment and this calculation, no support is necessary. <u>(SER 92-014-01)</u>

2. <u>TM 93-003 (Common)</u>, Emergency Diesel Generator (EDG). TM 93-003 installs temporary jumper cable to bypass broken G01 EDG excitation potential transformer fuseholder disconnect contact. The "A" phase fuseholder contact for the G01 excitation potential transformer (EPT) was found broken. No replacement part was immediately available. This temporary jumper ensures solid connection between the downstream side of the fuseholder and the EPT.

   <u>Summary of Safety Evaluation</u>: The old spring clip was designed to disconnect the high and low sides of the fuseholder from the G01 potential transformer circuit when the fuse door was opened. This is a personnel safety feature of the fuseholder which does not compromise the temporary modification since the excitation breaker must be open before the fuse door can be opened.

   A bolted jumper connection provides better contact than the existing spring clip arrangement. The No. 6 AWG bonding jumper is adequately protected by the 30 amp potential transformer fuse, based on the 1990 National Electrical Code. Note this cable size is equivalent to the existing feeder cable size, and is therefore, adequate for the intended purpose.

   The length and arrangement of the jumper preclude it from interfering with other control cabinet components during a seismic event. Based on engineering judgement, the bolted bonding jumper would perform better than the spring contact during a seismic event.

All cable and lugs are QA released for use in the diesel generator control cabinets. Independent verification is used for ensuring the jumper is properly installed and removed when replacement spring contact(s) are available.

No Appendix R considerations exist due to the jumper being contained within the G01 control cabinet. (SER 93-016)

3. TM 93-004 (Common), Emergency Diesel Generator (EDG). Fuse F2 is removed from C34 (G01 control cabinet). The removal of this fuse results in the isolation of the primary side of PT-2. The secondary side of PT-2 is not altered. PT-1, PT-2, and PT-3 are set up in a WYE-WYE configuration.

Summary of Safety Evaluation: The removal of fuse F2 isolates the primary side of PT-2 from the generator output. The secondary side of PT-2 was never connected and is isolated. If a fault occurred on PT-2 with F2 removed, it would cause the neutral of the WYE to become grounded on the PTs. This does not effect the ungrounded WYE connection of G01, therefore does not affect the operation, protection, or failure modes of G01. With G01 unaffected by this change, the probability of an accident, or the consequences of an accident does not increase. (SER 93-017)

4. TM 93-006 (Common), Emergency Diesel Generator (EDG). The corrosion inhibitor in G01 and G02 EDGs was changed from glycol based solution to borate-nitrate. This resulted in an expenditure of nitrates as they form the oxide layer and drop in the pH level in G02. The old oxide layer is now circulating in G02 in the form of soluble and insoluble iron. TM 93-006 removes the insoluble iron from solution by filtering the coolant and at the same time adds borate-nitrate to the upper end of the control limit (1100 ppm).

Summary of Safety Evaluation: The temporary modification controls a work plan intended to be worked while G02 is in service. During the conduct of this recirculation, the design, operation or function of G02 is not altered. G02 is fully capable of performing its safety-related function if called upon without immediately aborting from this temporary modification.

In the event of an engine start, TM 93-006 requires an operator to stand by the engine coolant drain valve. The operator's job is to close the coolant drain valve and recap the expansion tank. This action is conservative as the recirculation line component is capable of operation at G02 coolant design (55 psi and 190°F) if the valve was not shut. The intent of the closure is to limit the coolant bypass flow and isolate the unqualified portion of the system (install by the temporary modification). Although the temporary modification installation is not seismically analyzed, its installation would survive a DBE in view of the low acceleration involved at the EDG and the flexibility of the installation hoses. The installation does not negatively impact the seismic qualification of the existing components. This is based on judgement in view of the above and support configuration of the existing system. (SER 93-031)

5. TM 93-008 (Unit 1), Main Steam. TM 93-008 installs a blind flange in place of atmospheric steam dump valve, 1MS-2016. A 5", 600# blind, raised-face flange in place of valve 1MS-2016. Manual isolation valve, 1MS-227, will be used for controlling steam dumping to atmosphere if needed.

Summary of Safety Evaluation: The replacement of the valve with a blank flange does not cause a safety concern for the operation of the plant. A 600# rated flange is used which is the same as the original design of the valve. The flange will not see full system pressure as the differential pressure is at the upstream valve MS-227. The downstream pressure of this valve is vented to atmosphere. The flange is in place of the valve for personnel safety if the upstream valve needs to be operated. The flange does not pose an unreviewed safety question. Operation of the system can be accomplished through manual operation of the upstream valve. EOP-3.3 and EOP-3 can be performed using the manual valve as the EOP states that you are to use any available means to dump steam. The reason for needed operation of the pipe and flow path is to allow cooldown/depressurization using the intact steam generator following a steam generator tube rupture without condenser steam dumps available. (SER 93-042)

6. TM 93-021 (Unit 2), Main Steam. TM 93-021 installs a blind flange in place of atmospheric steam dump valve, 2MS-2016. A 5", 600# blind, raised-face flange in place of valve 2MS-2016. The manual isolation valve is used for controlling steam dumping to atmosphere if needed.

   Summary of Safety Evaluation: The replacement of the valve with a blank flange does not cause a safety concern for the operation of the plant. The flange is a 600# rated flange which is the same as the original design of the valve. The flange does not see full steam pressure as the differential pressure is at the upstream valve. The downstream pressure of this valve is vented to atmosphere. The flange is in place of the valve for personnel safety if the upstream valve has to be operated. The flange does not pose an unreviewed safety question. Operation of the system can be accomplished through manual operation of the upstream valve. EOP-3.3 and EOP-3 can be performed using the manual valve as the EOP states that you are to use any available means to dump steam. The reason for needed operation of the pipe and flow path is to allow cooldown/depressurization using the intact steam generator following a steam generator tube rupture without condenser steam dumps available. (SER 93-051)

7. TM 92-041 (Unit 1) and TM 92-042 (Unit 2), 4160 V. TMs 92-041/042 address the installation of modifications E-124 and E-125 that caused the condensate pump breakers to lose their anti-pumping feature during the condition of low steam generator feed pump suction pressure and undervoltage on A01/A02. The anti-pumping feature is lost due to the installation of a "B" contact in series with the "X" and "Y" relay coils in the control circuit.

   Summary of Safety Evaluation: The TMs bypass the "B" contacts in the condensate pump closing circuits to ensure breaker pumping does not occur. Doing this may cause the X and/or Y relays to repeatedly energize while the breaker is closed if the steam generator feed pump suction pressure drops below the PS-2141 setpoint of 181 psi. Review of the control room shift log indicates that recent steam generator feed pump suction pressures are normally maintained above 220 psi. While the TMs are installed, a voltage counter was connected across PS-2141 to determine if steam generator feed pump suction pressure occasionally dropped below 181 psi and energized the "X" and/or "Y" relays. (SER 93-003)

8. TM 93-030, (Unit 1), Main Steam. TM 93-030 installs a blind flange in place of valve bonnet on atmospheric steam dump valve 1MS-2015. A 5", 600# blind, raised-face flange to be put in place of a valve bonnet on 1MS-2015. The manual isolation valve will be used for controlling steam dumping to atmosphere if need be.

   Summary of Safety Evaluation: The replacement of the valve with a blind flange does not cause a safety concern in the operation of the plant. The flange is a 600# rated flange which is the same as the original design of the valve. The flange will not see full system pressure as the differential pressure is at the upstream valve. The downstream pressure of this valve is vented to atmosphere. The flange is in place of the valve for personnel safety if the upstream valve has to be operated. The flange does not propose an unreviewed safety question. Operation of the system can be accomplished through manual operation of the upstream valve. EOP-3.3 and EOP-3 can be performed using the manual valve as the EOP states that you are to use any available means to dump steam. The reason for needed operation of the pipe and flow path is to allow cooldown/depressurization using a steam generator. (SER 93-078)

Summary of Safety Evaluation: This is a revision to SER 93-078. A 5", 600# blind raised-face flange are to be put in place of valve bonnets on 1MS-2015, 2MS-2015, 1MS-2016 and 2MS-2016. The reason for needed operation of the pipe and flow path is to allow cooldown/depressurization using a steam generator. Use of the valve will be controlled by issuing a red tag series to the duty shift superintendent (DSS) to tag the manual isolation valve shut. The manual isolation valve will only be used if considered necessary and appropriate by the DSS. The probability of the valve being stuck open when being used is very small. This occurrence is judged to be similar to a stuck open code safety valve which is considered in the FSAR and is bounded by the main steam line break accident. Emergency operating procedures exist for mitigating such an occurrence. (SER 93-078-01)

SPARE PARTS EQUIVALENCY EVALUATION DOCUMENTS (SPEEDs)

1. SPEED 93-012, Replacement of Discs for 1CV-350. The disc material changes from ASTM A296 CF8 to ASTM A479 TP304 Cond A. A479 TP304 Cond A is comparable to A296 CF8 in material properties. Hardfacing is changed from Stellite to Deloro 40. The male and female discs are produced with an extra 0.050" facing. Discs may require machining for leak-tight fit. Other dimensions remain the same. Discs are made to be direct replacements. Similar work was performed on 2CV-350 (SPEED 92-096).

   Summary of Safety Evaluation: The male and female discs in 1CV-350 gate valves are replaced due to the original discs being warped. The new male and female discs are made out of equivalent materials. Both materials have similar heat treatments. The chromium and nickel compositions are essentially the same, which gives similar corrosion resistance. There is a small difference in the allowance for silicon, but it is negligible. The A479 specification is a general specification for raw material fabrication, whereas the A296 specification is a specification for the fabrication of a cast component.

   Although the final products of these specifications are different, the materials are essentially the same. The use of the ASTM A479 TP304 material is, therefore acceptable for manufacture of the replacement valve disc.

   The original valve disc was hardfaced with Stellite 6 as directed by specification. Stellite hardfacing is, however being eliminated from primary systems to minimize activation products and contaminants within the reactor coolant system. The new discs have a Deloro 40 (a nickel alloy) hardfacing, which is comparable to the stellite for this application.

   The manufacture of the new male and female disc satisfy the intent of the original specification and other codes in effect at the time of valve manufacture. The new valve disc is also manufactured in accordance with the current edition of the manufacturer's QA program that conforms to ASME Code Section III. The new disc is, therefore equivalent to the original and acceptable for use. (SER 93-015)

2. SPEED 93-089, Insulation Material Change For Containment Fire Breaks. The SPEED describes a change in materials for 13 containment fire breaks. The fire breaks are located at the floor level of the respective elevations: one at El 66' beneath PP-6; ten on the El 46' at trays 2W102, 2VB04, 2VD04, 2VJ02, 2VT02, 2VM02, 2VW02 and two unlabeled trays; two on the El 21' at 2VB02 and one unlabeled tray. The existing material is ceramic fiber and the new material is Dow Corning 3-6548 RTV silicone foam.

   Summary of Safety Evaluation: Currently in containment, fire breaks are constructed of ceramic blanket material (fibrous insulation), held in place with mastic coating. The fire breaks are not engineered fire protection features identified in either the FSAR or the Fire

Protection Evaluation Report. NRC Bulletin 93-02 and RG 1.82 limit the use of fibrous insulation materials inside of containment due to the potential impact of clogging sump screens during the accident phase of sump recirculation. While the ceramic blanket has excellent heat resistance characteristics, the material has no tensile strength. This is an indication that it will not maintain against a water head.

Dow Corning 3-6548, is the current material utilized in fire barrier penetrations throughout the plant. It has a satisfactory rating during ASTM D-E119 hose stream test. It is a material qualified for the containment accident environments and is currently used in containment fire stops. Tests concluded that the foam has a static pressure head capability of 5 psi. This is superior to capabilities of ceramic blanket material. In its application in containment, the foam product 3-6548 is installed with a nominal width of 8" to which is equivalent to a two hour fire rated seal thickness installed in the floor penetrations as a fire break. (SER 93-081)

MISCELLANEOUS EVALUATIONS

1. FSAR Section 5.2, Containment Penetration Classifications.

In the detailed design of the nuclear plant systems, certain lines required minor modification to the arrangements defined by these classes in order to implement the basic redundant barrier criterion.

Table 5.2.1, Note D: The designation "Special" indicates that the line cannot be classified in accordance with the five general classifications. In these lines, special arrangements of isolation features provide the redundant barriers and are described in the note associated with each figure.

Summary of Safety Evaluation: The FSAR changes involve the following:

- Administratively change the FSAR classification for containment penetrations 52-1, 52-2, 53-1, 53-2 and X-1 from Special to Class 5.
- Administratively change the FSAR classification for containment penetrations 69 and 70 from Class 2 to Special.
- Administratively change the FSAR classification for containment penetrations 13, 22, and 27 from Class 3 to Special.

  Changing the class designation for these penetrations results in the removal of four motor-operated valves (MOVs) and six manual valves designated as containment isolation valves (CIVs) from the FSAR.

- Remove the CIV classification from 22 manual valves. The penetrations associated with these valves are classified Special and require no revision to their classification.

The manual valves currently designated as CIVs in the FSAR to be reclassified as non-containment isolation valves are not relied upon under accident conditions to effect containment isolation, nor are any shut by procedure to achieve containment isolation. Similarly, the MOVs currently designated as CIVs in the FSAR to be reclassified as non-containment isolation valves are not relied upon under accident conditions to effect containment isolation, are not shut by procedure to achieve containment isolation, and do not receive a containment isolation signal.

None of these valves are required to be Type C tested under 10 CFR 50, Appendix J, nor is it appropriate to do so. None of the valves have ever been relied upon by procedure or as a matter of operational practice to effect containment isolation means. It is improper to credit any of them as being part of the margin of safety associated with the isolation of their respective containment penetrations.

The changes do not result in any physical changes to any plant system, structure, or component. No plant procedure requires revision. Redesignating a penetration from one class to another is a change which is administrative in nature. (SER 89-073-01)

2. FSAR Section 5.2, Class 1, Containment Penetration Definition.

FSAR Class 1 (outgoing lines) is defined as normally operated outgoing lines connected to the reactor coolant system (RCS) are provided with at least one automatically operated trip valve and one manual isolation valve in series located outside containment. In addition to the isolation valves, each line connected to the RCS is provided with a remote operated root valve located near its connection to the RCS.

The revised definition is normally operated outgoing lines connected to the RCS are provided with two automatic trip valves in series, one located inside containment and one located outside containment.

Summary of Safety Evaluation: The FSAR changes involve the following:

- Change the FSAR definition for Class 1 containment penetrations as discussed above.
- Classify six automatic trip valves as containment isolation valves (CIVs).
- Remove "containment isolation valve" designation from 18 manual valves that do not establish a containment isolation barrier in a post-accident scenario.

The six automatic trip valves being added to the FSAR as CIVs receive a containment isolation signal and function to effect containment isolation under accident conditions. All six valves are Type C tested for leak tightness under 10 CFR 50, Appendix J. It is appropriate that these valves be classified as CIVs since they establish a containment isolation barrier. With the exception of SC-955, which is located inside the missile barrier, all valves meet the design requirements necessary to be classified as CIVs. The matter of SC-955 missile protection is addressed via condition report 93-278.

Conversely, the 18 manual valves designated as CIVs in the FSAR do not function automatically to effect containment isolation, nor are any shut by procedure to effect containment isolation. None of these manual valves are Type C tested under 10 CFR 50, Appendix J, nor would it be appropriate to do so. The fact that none of these valves are relied upon to establish a containment isolation barrier means that it is improper to credit any of them as being part of the margin of safety associated with the isolation of their respective containment penetrations. (SER 89-073-02)

3. FSAR Section 5.2, Class 2, Containment Penetration Definition.

FSAR Class 2 (outgoing lines) is defined as normally operated outgoing lines not connected to the reactor coolant system (RCS) and not protected from missiles throughout their length are provided with at least one automatically operated trip valve or one remotely operated stop valve located outside the containment. Manual isolation valves in series with the trip or remote operated valves are also provided outside the containment unless the line is connected to a closed system outside containment.

The revised definition is normally operated outgoing lines not connected to the RCS and not protected from missiles throughout their length are provided with either two automatic trip valves in series or a closed system outside containment and either a remotely operated stop valve or an automatic trip valve in series.

Summary of Safety Evaluation: The FSAR change involves revising the definition for Class 2 containment penetrations as described above and administrative in nature only. No system, structure, or component is physically altered or modified, nor do any plant procedures require revision. The designation of valves as containment isolation valves is not being altered as a result of this change to the FSAR Class 2 penetration definition. No component design basis is being changed. (SER 89-073-03)

4. FSAR Section 5.2, Class 3, Containment Penetration Definition.

The FSAR Class 3 (incoming lines) is defined as incoming lines connected to open systems outside containment are provided with two automatically operated trip valves in series. (A check valve is used in lieu of one of the trip valves in certain lines). One of these valves may be located inside containment between the reactor coolant system (RCS) shield walls and the containment wall.

Incoming lines connected to closed systems outside containment are provided with at least one check valve located either inside or outside containment. A manual or remotely operated isolation valve in series with the check valve is also provided outside containment.

The revised definition is incoming lines connected to open systems outside containment are provided with two automatic trip valves in series, one of which may be located inside containment. Incoming lines connected to closed systems outside containment are provided with one automatic trip valve located inside containment.

Summary of Safety Evaluation: The FSAR changes involve the following:

- Change the definition for Class 3 containment penetrations as discussed above.
- Remove the containment isolation valve (CIV) classification from four motor-operated valves (MOVs) and twelve manual valves.

  The component cooling water and charging system penetrations are qualified closed systems outside containment. This established an additional containment isolation barrier for penetrations 15, 16, 19, 26, 29a, 29b, and 32c.

  The eight valves associated with penetrations 26, 29a, and 29b are simple manual valves. The eight manual valves are not Type C leak tested under 10 CFR 50, Appendix J, nor is it required or appropriate to do so.

  The four valves associated with penetrations 19 and 32c are also simple manual valves. The four manual valves are Type C leak tested under 10 CFR 50, Appendix J. However, this testing is not required and discontinuation is consistent with the removal of the CIV classification.

  The four valves associated with containment penetrations 15 and 16 are MOVs. These four valves are Type C leak tested under 10 CFR 50, Appendix J. However, this testing is not required and discontinuation is consistent with the removal of the CIV classification.

  The valves in question do not function automatically to effect containment isolation, nor are they procedurally under accident conditions to effect containment isolation.

The four MOVs and twelve manual valves designated as CIVs in the FSAR, are to be declassified, are not relied upon to establish one of two required containment isolation barriers under accident conditions. None of these valves are required to be Type C tested under 10 CFR 50, Appendix J, nor would it be appropriate to do so. The fact that none of these valves are relied upon to effect containment isolation means that it is improper to credit any of them as being part of the margin of safety associated with the isolation of their respective containment penetrations. (SER 89-073-04)

5. MWR 925771, Main Steam System. The MWR abandons the freeze protection circuit on the Unit 2 "B" main steam line within the facade.

Summary of Safety Evaluation: U2R18 primary inservice inspection (ISI) (MWR 924188) required that the asbestos insulation be removed from 2MS-2017, the Unit 2 "B" main steam isolation valve (MSIV). During asbestos abatement, we discovered that the condition of the main steam line's facade freeze protection circuit was very poor. Therefore, MWR 925771 abandons the freeze protection circuit (2FF20A).

The main steam header freeze protection purpose is not precisely known. However, the most likely reason for its original installation was to assist in starting the plant from a "cold iron" condition in extremely cold weather. For example, we may have a steam generator in wet lay-up, or desire to perform a hydrostatic test of a steam generator. Either of these situations would require that the main steam line within the facade be filled with water. The possibility exists that some freezing could occur on the inside surface of the pipe if either of these situations were to arise during cold weather. Another possible explanation for the existence of the freeze protection circuit is a desire to keep the pipe warm for pressurizing (brittle-fracture concerns). Other options exist to address these situations (i.e., during cold conditions, another heat source could be rigged to prevent freezing within the main steam line).

The postulated negative effects from abandoning this freeze protection circuit deal with subcritical reactor conditions. Because of this, failure is limited to the main steam system during periods when the steam generators are not in service. The results of such effects would be bounded by the steam line break accident, which is analyzed in FSAR Chapter 14. In addition, the main steam lines would be drained and depressurized before the steam line's temperature would approach the freezing point during an actual accident. Therefore, abandoning this freeze protection circuit has no impact on any postulated accident condition.

The only licensing-basis document that mentions the heat tracing on the main steam line is FSAR, Figure 10.2-1A. This figure will be revised with the annual FSAR update. (SER 93-056)

6. MWR 925818, Inspection and Possible Replacement CV-350 Discs. The MWR work plan unbolts and separates the bonnet from the body of the emergency borate valve, 1CV-350. This inspects the discs which are believed to be warped. Replacement parts are available if existing parts do not conform to standards. This work plan is executed in conjunction with MR 91-133*B during U1R20.

Summary of Safety Evaluation: The MWR removes the valve bonnet from the body of 1CV-350 along with the valve disc and stem. The valve maintenance is performed in accordance with the component instruction manual. When the valve is placed back in service, the replaced parts have no effect on the operation of the boric acid system. The post-maintenance testing is accomplished with the motor-operated valve diagnostic system and does not impact the operation of the system. The interim conditions during U1R20 isolates the system to allow it to be flushed and drained, and the heat tracing deenergized. In this condition, it is not possible to add concentrated boric acid using the normal boration path. A boration flow path from the boric acid storage tank or refueling water storage tank (RWST) to the reactor coolant system through the safety injection pumps or the RWST to the reactor coolant system through the residual heat removal pumps exists at all times in accordance with technical specification requirements. (SER 93-008)

7. MWRs 926271 and 926272, SMPs 1130, 1130-1, 1130-2, 1130-3 and 1130-4, As-Built Wire Tracing of G02 Emergency Diesel Generator (EDG) Control Circuitry in Main Control Board (MCB) Section C02. The MCB as-built project continues with hand-over-hand wire tracing of the diesel generator, G02, circuitry within Section C02.

Summary of Safety Evaluation: The work is performed while G02 EDG is out of service for its annual inspection. Therefore, this evaluation does not address removal of G02 from service, but rather looks at safety concerns associated with the actual as-built activity.

The as-builting process begins with a visual inspection of the terminal blocks on Riser 37. The number of wires at each terminal is recorded in a wire tracing book so that the configuration after wire tracing can be compared to the original configuration. This ensures that any wires that might accidently be lifted are detected.

MCB Section C02 contains common electrical controls which includes the following: Unit 1 & 2 turbine generators, 345 kV, 13.8 kV, Unit 1 & 2 4160 V, Unit 1 & 2 480 V, emergency diesel generators, and gas turbine. Only G02-related components have been selected to be traced; this prevents other systems from being directly affected. Furthermore, with the exception of four 480 V breakers, these systems are not indirectly affected because they do not have wires in Riser 37 and are physically separated from the G02 circuitry.

The G02 metering in C02 is tested by opening sliders and test switches, hooking up sources to these points and then generating current and voltage to check out the meters. Concurrent checks are done before the sliders and test switches are opened to ensure the electricians are at the proper location and prevent any accidental actuation of components as a result of the action. Independent checks are done to ensure the sliders and test switches are closed after testing.

Breaker 2B52-26C is tested by taking voltage readings. An MWR closely monitors the functions of G02 metering and controls in C02 during TS-2.

Follow-up testing is performed upon completion of wire tracing. This is done on all G02 circuitry in the MCB and on all other safety-related circuitry in Riser 37. Also, the terminal blocks are visually inspected before and after wire tracing to ensure that the original configuration has not been changed. (SER 93-004)

8. MWRs 9300695, 930696, and SMPs 1133, 1133-1, 1133-2 and 1133-3, As-Built Wire Tracing of G01 Emergency Diesel Generator (EDG) Control Circuitry in Main Control Board (MCB) Section C02. The main control board as-built project continues with hand-over-hand wire tracing of G01 EDG circuitry within MCB C02. This occurs during the G01 annual inspection.

Summary of Safety Evaluation: The as-builting process begins with a visual inspection of terminal blocks on Riser 34. The number of wires at each terminal is recorded in a wire tracing book so a comparison of the configuration after wire tracing can be compared to the original configuration. This ensures that any wires that might accidently be lifted are detected.

MCB C02 contains common electrical controls which includes the following: Unit 1 & 2 turbine generators, 345 kV, 13.8 kV, Unit 1 & 2 4160 V, Unit 1 & 2 480 V, emergency diesel generators, and gas turbine. Only G01-related components are selected to be traced;

this prevents other systems from being directly affected. All traced circuitry is tested upon restoration via SMPs 1133-2 and 1133-3.

The G01 EDG metering in C02 is tested by opening sliders and test switches, hooking up sources to these points and then generating current and voltage to check out the meters. Concurrent checks are completed before the sliders and test switches are opened to ensure the electricians are at the proper location and prevents an accidental actuation of components as a result of the action. Independent checks are performed to ensure the sliders and test switches are closed after testing. An MWR closely monitors the functions of G01 metering and controls in C02 during performance of TS-1.

In summary, the G01 circuitry is traced while G01 EDG is out of service for its annual inspection. Components and risers involved are identified prior to wire tracing, preventing other systems from being affected. G01 circuitry is tested upon restoration. (SER 93-007)

9. MWRs 931746, 931747, 931748, 931749, STPT 2.5 and FSAR 8.2, FSAR 8.2 Containment Fan Cooler Time Delay Relay Settings. The evaluation addresses changing the containment accident fan cooler time delay relay setpoints in FSAR 8.2 to correspond to the requirements of FSAR Table 14.3.2-1 and FSAR Page 14.3.4-14.

   Summary of Safety Evaluation: The present fan time delay settings installed in Unit 2 potentially do not agree with the requirements for maximum delay of 60 seconds prior to heat removal initiation, shown in FSAR 14.3.4-14. The time delay settings installed in Unit 1 do not allow for a minimum of 35 seconds from time of accident initiation until time of fan cooler starting, as required by FSAR Table 14.3.2-1. The evaluation also addresses the changes to the setpoint document, and the work control documents used to install these setpoint changes. The work control documents require placing Train A residual heat removal (RHR) pump control switch in pullout while the timing is set, to prevent inadvertent starting of the RHR pump during the test. This may require the unit entering the LCO for this condition. Setting the time delay relays requires actuating them, which would prevent them from performing their design function. If required by the plant condition, a dedicated operator is stationed to sequence the containment fan coolers on as required in the highly unlikely event of an accident during the testing. This allows the containment fan coolers to be considered operable throughout this procedure.

   New setpoint values and FSAR expressions were determined for TDR-16, 17, 26, and 27. The calculation incorporated all relevant FSAR input, manufacturer's data, diesel loading information, and plant specific information. The acceptable range values of FSAR 8.2 will be changed to reflect these new values. Setting the time delay relays in accordance with the FSAR requirements does not involve an unreviewed safety question. The work plan used to set the relays requires that a unit enter the appropriate LCO for having an RHR pump out of service during the time that its control switch is in pullout. This aspect is covered in the work control documents.

   Setting the containment fan cooler time delay relays to comply with the accident analysis does not involve an unreviewed safety question. The interim conditions required to implement these settings may place the plant in an LCO. This condition has been previously evaluated in the FSAR, and does not involve an unreviewed safety question. (SER 92-089-01)

10. MWR 932847, Instrument Bus Transfer Test. MWR 93-2847 tests different methods of shifting Unit 2 instrument buses while monitoring instrument bus voltage and inverter output voltage. The MWR will determine the method for shifting instrument buses that causes the least amount of instrument bus voltage transients.

Summary of Safety Evaluation: The present method of shifting instrument buses often results in numerous control room alarms due to momentary loss of bus voltage during shifting. During the test, instrument buses 2Y-01, 2Y-101, 2Y-03 and 2Y-103 are shifted between the normal and alternate source while the static transfer switches for the inverters are in the "backup source" and the "inverter" positions.

The MWR is performed during U2R18 with the core defueled to reduce the risk associated with instrument bus shifting. Testing is coordinated with other outage activities so that potential instrument bus voltage transients does not affect any system important to plant safety. The worst case occurrence during the test would be the deenergization of a Unit 2 instrument bus. Power to the affected bus could quickly be restored to provide the necessary indication and control functions. This test does not affect any Unit 1 instrument buses.

Instrument bus breaker mechanical interlocks are disabled to allow parallelling of normal and alternate sources for a short time during instrument bus shifting. This is performed via a procedural temporary modification with the interlocks restored following the test. The inverters affected by the test are verified as being in sync prior to instrument bus shifting. (SER 93-080)

11. MWRs 934093, 934094, 935946 and 935947 , Component Cooling (CC) Heat Exchanger (HX) Coating Installation. The MWRs involve the replacement of service water (SW) side epoxy coating in the CC heat exchangers, HX-12A, HX-12B, HX-12C and HX-12D. Replacement of the degraded coating helps to mitigate any corrosion present and ensure the reliability of the affected HXs operation.

Summary of Safety Evaluation: The coating installation project affects the CC system, described in FSAR Section 9.3 and the SW system, described in FSAR Section 9.6.2.

There are four CC HXs with one HX in service for each unit, and two idle HXs serving as standby for either unit. This allows for the removal of one HX from service for maintenance or in case of failure, without degrading the normal and emergency cooling capability of the CC system for either operating unit.

The CC HXs are QA safety-related components. The coating material is evaluated to meet all associated QA requirements. The material is suitable for application, taking into account the effects of temperature, pressure, radiation, water chemistry, etc. Heat transfer capability is not affected because the coating is only applied to the SW side of the waterboxes and tubesheets, not the tubes.

The probability of a coating failure is small; however, if it did occur, system design and redundancy would prevent adverse consequences. The coating surfaces are visually inspected for degradation during HX maintenance outages. Inspection of surfaces prior to application of the coating and regular maintenance inspections should prevent catastrophic failure of the coating and allow for normal erosion of the material.

No unreviewed safety questions exist related to the coating installation. No fission product barriers are affected. (SERs 93-005 and 93-055-01)

12. SQUG Methodology, Implementation of the SQUG Methodology for the Seismic Design and Verification of Modified, New, and Replacement Equipment. The change revises FSAR Appendix A to allow the use of seismic experience data for the seismic design and verification (following installation) of modified, new, and replacement equipment classified as Seismic Class I. The FSAR does not currently specify a method to be used for verifying or qualifying the design and installation of Seismic Class I equipment for seismic adequacy.

Summary of Safety Evaluation: The NRC approves the SQUG methodology as an acceptable seismic verification methodology for USI A-46 plants such as Point Beach. The NRC SSER #2 also states that "the criteria and procedures described herein are determined to be an acceptable evaluation method for verifying the seismic adequacy of the equipment in USI A-46 plants including future modifications and replacement equipment in these plants." This same reference also states that the SQUG methodology is "determined to be an acceptable evaluation method for verifying the seismic adequacy of new equipment in USI A-46 plants." TACs M69472 and M69473 staff states that the "staff recognizes that you may revise the licensing basis in accordance with 10 CFR 50.59 to reflect the acceptability of the USI A-46 (GIP) methodology for verifying the seismic adequacy of electrical and mechanical equipment covered by the GIP." Therefore, it is clear that the NRC has approved the SQUG methodology for use on modified, new, and replacement equipment as the proposed FSAR change allows. The only exceptions are as noted in the FSAR change that for new installations and newly designed anchorages, the factor of safety currently recommended for new nuclear plants in determining the allowable anchorage loads shall be met and that the SQUG methodology will not be used for the design of new tanks. (SER 93-014)

13. Technical Specification 15.4.3, Basis Change for TS 15.4.3, "Primary System Testing Following Opening."

Summary of Safety Evaluation: The basis for TS 15.4.3, "Primary System Testing Following Opening," states:

"Repairs on components 2" diameter or smaller are less significant to safety since the basis of design is that the charging pumps can accommodate a break of this magnitude. Pipes of this size generally employ socket-weld techniques, therefore only visual and surface inspection techniques are applicable..."

Calculation N-90-015 demonstrates that at least two charging pumps are required to maintain reactor coolant inventory long enough to preclude safety injection at no-load conditions with a 3/8" ID break. In addition, the 2" diameter delineated in TS 15.4.3.b and 15.4.3.c relates to the physical limitations of the NDE methods involved rather than charging pump capacity. Therefore, the basis for TS Section 15.4.3 is changed to remove the association of charging pump capacity to NDE method as follows:

> "Components of 2" diameter or smaller generally employ socket-weld techniques. Applicable inspection techniques for socket welds are visual and surface examinations."

An unreviewed safety question does not exist. This technical specification basis change does not affect its associated technical specification; therefore, a TS change is not required. Since TS Bases may be changed in accordance with 10 CFR 50.59, this safety evaluation is sufficient authorization to complete the change. (SER 92-106)

14. U1C21, Point Beach Unit 1 Cycle 21 Fuel Reload. The Unit 1 Cycle 21 reload contains 12 fresh Region 23A upgraded optimized fuel assemblies (OFA) at 3.6 w/o, 16 fresh Region 23B upgraded OFA at 4.0 w/o, 12 Region 22A upgraded OFA, 16 Region 22B upgraded OFA, 12 Region 21A upgraded OFA, 16 Region 21B upgraded OFA, 16 Region 20A upgraded OFA, 12 Region 20B upgraded OFA, 8 Region 19A upgraded OFA, and 1 Region 12 Unit 2 standard fuel assembly. The U1C21 core is the fifth reload containing a full region of upgraded OFA fuel.

Summary of Safety Evaluation: The licensing basis accidents which potentially could be affected by the reload core design were reviewed. The core design was performed assuming the reactor coolant system can be operated at a pressure of either 2000 or 2250 psia. The Cycle 21 design does not cause previously acceptable safety limits to be exceeded, provided that:

- Cycle 20 burnup is bounded by 9400 and 10600 MWD/MTU. Actual Cycle 20 burnup was 10,117 MWD/MTU;

- Cycle 21 burnup is limited to the end-of-full-power-capability (EOFPC, which is defined as the burnup of fuel when all control rods are fully withdrawn, and less than or equal to 10 ppm of residual boron at the Cycle 21 rated power condition of 1518.5 MWt) plus 1500 MWD/MTU power coastdown;

- There is adherence to the plant operating limitations given in the Technical Specifications;

- The safety aspects on the reactor internals of utilizing PPSAs have been assumed by WE; and

- The effect of the Cycle 21 design of the boron dilution event in cold shutdown has been assumed by WE. (SER 93-032)

15. U2C20, Point Beach Unit 2 Cycle 20 Fuel Reload. The reload core involves a potential change to the facility or its operation as described in the FSAR. For the Cycle 20 core, the minimum measured flow rate was reduced by 2600 gpm, to 179,200 gpm.

Summary of Safety Evaluation: The licensing basis accidents which potentially could be affected by the reload core design and the flow reduction were reviewed. Reanalyses were performed prior to the start of Unit 2 Cycle 20 design. The reanalyses show the acceptability of revised core thermal limits, new loss-of-flow statepoint, new locked rotor statepoint and new dropped rod limit lines. The reanalyses are documented in letter WEP-93-594, dated April 30, 1993. The evaluations were performed using methodology, models and procedures which the NRC staff has reviewed and approved.

As a result of the Cycle 20 evaluation, it is concluded that the Cycle 20 design does not cause safety limits to be exceeded, provided that the following conditions are met: Cycle 19 burnup is bounded by 10200 and 11200 MWD/MTU (Final Cycle 19 burnup is 10606 MWD/MTU); Cycle 20 burnup is limited to the end-of-full-power-capability (EOFPC) which is defined as the burnup of fuel when all control rods are fully withdrawn, and less than or equal to 10 ppm of boric acid at the Cycle 20 rated power condition of 1518.5 MWt, plus 1500 MWD/MTU power coastdown operation; there is adherence to the plant operating limitations given in the TS; TS Change Request 160 has received approval prior to the commencement of Unit 2 Cycle 20 operations; the safety aspects on the reactor internals of utilizing peripheral power suppression assemblies (PPSAs) have been evaluated; and the effect of the Cycle 20 design of the boron dilution event in cold shutdown has been evaluated; the reactor coolant system is operated at a nominal pressure of 2000 psia. (SER 93-083)

V. NUMBER OF PERSONNEL AND PERSON-REM BY WORK GROUP AND JOB FUNCTION - 1993

Job Group Station Employees	Total rem for Job Group	Work Function and Total Person-rem					
		Reactor Operations & Surveillance	Routine Maintenance	Inspections	Special Maintenance	Waste Processing	Refueling
Operations	23.050	13.250	-------	4.700	-------	0.400	4.700
Maintenance	45.280	-------	26.050	1.010	-------	-------	18.220
Chemistry & Health Physics	15.920	13.440	-------	-------	-------	2.480	-------
Instrumentation & Control	2.020	-------	1.080	0.010	-------	-------	0.930
Administration & Engineering, Regulatory Services	2.630	0.220	-------	2.410	-------	-------	-------
Utility Employees	23.710	2.770	19.530	1.410	-------	-------	-------
Contractor Workers & Others	73.563	0.240	-------	31.273	39.070	2.980	-------
GRAND TOTALS	186.173	29.920	46.660	40.813	39.070	5.860	23.850

1257 individuals were monitored exempt from the provisions of 10 CFR 20.

NUMBER OF PERSONNEL GREATER THAN 100 MREM BY GROUP AND JOB FUNCTION - 1993

Work & Job Function	Number of Employees Greater Than 100 mrem			
	Station Employees	Utility Employees	Contract Workers and Others	Totals
Reactor Operations & Surveillance	97	9	1	107
Routine Maintenance	121	19	-------	140
Inpections	127	3	57	187
Special Maintenance	-------	------	107	107
Waste Processing	96	-------	4	100
Refueling	121	-------	------	121
TOTALS	562	31	169	762

Workers may be counted in more than one category. Numbers are total numbers of individuals.

VI. STEAM GENERATOR EDDY CURRENT TESTING

UNIT 1

Inspection Plan: During the Unit 1 Refueling 20 outage, eddy current testing was performed from April 15, 1993, to April 17, 1993. An approximate 20% sample was inspected full length and an additional first support sample was done around the peripheral to address loose parts concerns. The extent tested in each steam generator is as follows:

Eddy Current Inspection Plan		
Extent of Inspection	Number of Tubes	
	"A" SG	"B" SG
Full Length	588	591
No. 6 TSP	55	55
No. 1 TSP	202	206
Totals	845	852

Inspection Results: The results of these inspections revealed 137 tubes in the "A" steam generator with reportable indications, and 47 in the "B" steam generator. The following is a summary of the eddy current inspection results listing the largest reportable indication per tube:

Eddy Current Inspection Results Hot Leg (Cold Leg)		
	"A" SG	"B" SG
MBM	78 (66)	70 (52)
MMB	8 (8)	8 (7)
20-29%	1 (2)	1 (0)
		1 (0)
Totals	87 (76)	80 (59)

% - Percent Through Wall Indication
MBM - Manufacturing Burnishing Marks
MMB - Multiple MBMs

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

Tubes with Indications - Not Plugged: The following is a list of tubes which had indications but were not repaired or plugged as a result of eddy current testing during U1R20.

"A" Steam Generator Indications

1-6H or C - Tube Support Plate No. Hot or Cold Leg
BPH or C - Baffle Plate Hot or Cold Leg
MBM - Manufacturing Burnishing Marks
NQN - Non-Quantifiable Indication - Not Reportable
TSH or C - Tube Sheet Hot or Cold Leg

DNG - Ding
AV1-4 - Anti-Vibration Bar No.
MMB - Multipe MBMs

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
7-89	MBM	1C	14.9
8-7	MBM	2H	25.3
8-86	DNG	3H	27.5
9-17	MBM	4H	20.7
9-77	MBM	3C	48.1
9-77	MBM	BPC	17.7
9-81	MBM	TSC	7.1
9-83	MMB	4C	21.8
10-91	22	TSC	14.4
11-79	MBM	4C	30.4
11-79	MBM	5H	17.8
11-81	MMB	3C	24.9
12-6	MBM	1C	9.0
12-84	MBM	3C	14.9
12-84	MBM	3C	44.7
13-16	MBM	BPC	22.2
13-17	MBM	3C	47.3
13-74	MBM	2C	19.7
13-82	MBM	2C	10.9
13-86	MBM	2H	29.5
13-86	MBM	1C	45.9
14-4	DNG	TSH	2.5
14-4	DNG	TSH	1.3
14-9	DNG	5C	41.5
14-72	MBM	4H	48.0
14-73	MBM	2H	27.7

Row - Column	Indication	Location	Inch Mark
14-73	MBM	4H	14.9
14-73	MBM	BPH	7.3
14-73	MBM	2C	47.8
14-74	MBM	2H	25.6
14-76	MBM	4H	27.1
14-78	MBM	5C	42.1
14-78	MBM	4H	15.8
14-78	MBM	4H	49.4
14-80	MBM	3H	31.4
14-82	MBM	4C	41.2
14-83	MBM	BPC	13.3
14-83	MBM	5C	35.1
14-84	MBM	2H	15.3
15-6	MBM	3C	20.5
15-8	MMB	5H	16.7
15-8	MBM	5H	37.2
15-8	MBM	4H	22.8
15-8	MBM	1H	43.8
15-9	MBM	2H	5.9
15-10	MBM	3C	12.2
15-10	MBM	4H	15.0
15-14	MBM	4C	30.2
15-82	MBM	1H	10.4
16-73	MBM	BPH	24.2
16-73	MBM	1H	49.5
16-83	MBM	BPH	7.9
16-83	MMB	1H	20.0
16-88	DNG	BPC	9.7
16-89	26	TSC	14.9
17-7	MBM	2H	45.7
17-7	MBM	2H	45.8
17-11	MBM	1H	35.8

Row - Column	Indication	Location	Inch Mark
17-74	MBM	1H	45.1
17-80	MMB	4C	30.3
17-86	MBM	5H	21.8
18-7	MBM	2C	39.8
18-7	MBM	1C	8.6
18-9	MBM	3C	44.5
18-11	MBM	1C	5.3
18-12	MBM	5C	20.1
18-14	MBM	5H	37.0
18-16	MBM	2H	33.8
18-16	MBM	3C	19.2
18-86	MBM	5H	16.3
18-87	MBM	2H	43.2
19-9	MBM	3C	20.3
19-9	MBM	2C	46.8
19-78	MBM	BPH	8.2
19-83	MBM	1C	8.4
19-83	MBM	3C	3.1
20-17	MBM	1C	41.2
20-18	MBM	TSC	20.1
20-74	MBM	BPH	9.1
21-75	MBM	BPH	10.3
21-75	MBM	TSH	13.5
21-77	MBM	5H	27.9
21-87	MBM	TSH	4.6
22-9	MBM	3C	24.7
22-13	MBM	3C	32.7
23-11	MBM	2C	5.9
23-11	MBM	2H	34.0
23-11	MBM	1C	1.2
23-11	MBM	2H	22.1
23-11	MBM	3C	51.0

Row - Column	Indication	Location	Inch Mark
23-11	MBM	1C	45.1
23-12	MBM	5C	27.1
23-12	MBM	5C	19.0
23-12	MBM	1C	14.9
23-12	MBM	5C	22.0
23-12	MBM	4H	44.7
23-14	MBM	1H	26.5
23-14	MBM	5C	46.6
23-14	MBM	5H	25.9
23-14	MBM	BPC	6.4
24-15	MBM	3H	37.8
24-15	MBM	3H	35.4
24-15	MBM	5H	29.4
24-15	MMB	4C	32.8
24-85	MMB	BPH	22.3
25-11	MBM	4H	24.8
25-11	MBM	1C	8.8
25-12	MBM	2C	35.8
25-12	MBM	5C	3.0
25-12	MBM	4H	6.0
25-13	MMB	BPC	13.5
25-13	MBM	BPC	18.3
25-13	MBM	1C	22.7
25-13	MBM	4C	45.2
25-13	MMB	4H	13.2
25-13	MBM	BPC	21.7
25-14	MBM	4H	49.0
25-18	MBM	4C	45.5
25-18	MBM	BPC	14.0
25-18	MBM	3C	23.6
25-19	MBM	4C	24.9
25-19	MBM	3C	48.3

Row - Column	Indication	Location	Inch Mark
25-73	MBM	3H	25.9
25-79	MBM	5C	11.5
26-12	MMB	3C	19.4
26-12	MBM	1C	21.2
26-12	MBM	2H	2.6
26-13	MBM	BPH	22.6
26-14	MBM	3H	32.4
26-15	MBM	1H	43.2
26-15	MBM	1C	7.8
26-73	MBM	2H	20.7
26-75	MBM	TSH	9.0
26-76	MMB	BPC	18.5
26-76	MMB	3C	19.5
26-78	MBM	2H	42.6
26-79	MBM	AV2	16.1
26-79	MBM	4C	40.4
26-80	MBM	3H	49.0
26-80	MBM	1H	3.2
26-80	MBM	5C	19.8
26-82	MBM	4C	5.8
29-39	MBM	2H	22.8
29-40	MBM	4C	19.2
29-40	MBM	AV1	23.0
29-41	MBM	BPC	17.9
29-41	MBM	1C	44.3
29-42	MBM	3H	19.6
29-42	MBM	4H	32.7
29-43	MBM	4C	29.2
29-43	MBM	BPC	10.3
29-77	MBM	1H	6.4
30-16	MBM	2H	37.4
30-17	MBM	4C	28.6

Row - Column	Indication	Location	Inch Mark
30-25	MBM	1C	24.1
31-49	MBM	1H	20.5
31-49	MBM	1H	20.5
32-43	MBM	5C	20.4
32-61	MBM	5H	44.9
32-61	MBM	6H	72.4
32-67	MBM	BPH	13.9
32-69	MBM	4C	34.8
32-74	MBM	4H	24.8
32-75	MMB	2H	46.0
33-25	DNG	5C	38.7
33-29	MBM	2C	41.7
33-47	MBM	TSH	11.9
33-47	MBM	4C	19.3
33-47	MBM	2C	33.2
33-47	MBM	4H	32.2
34-21	MBM	BPC	4.1
34-24	MBM	3H	31.5
35-23	MBM	BPH	14.2
35-27	MBM	1H	23.9
35-42	MBM	1H	23.4
35-42	MBM	2C	42.2
35-42	MBM	1C	39.3
35-42	MBM	1C	44.3
35-45	MBM	2C	38.0
35-45	MBM	BPC	7.9
35-45	MBM	1C	33.0
35-47	MBM	3H	37.0
35-48	DNG	5H	39.5
35-48	NQN	5H	39.0
35-48	NQN	5H	43.9
35-48	NQN	5H	46.6

Row - Column	Indication	Location	Inch Mark
35-52	MBM	BPC	9.6
35-52	MBM	2H	41.1
35-53	MBM	3H	18.2
35-54	MBM	1H	32.4
35-55	MBM	2C	16.9
36-24	MMB	2H	24.2
36-24	MBM	2C	3.3
36-27	MBM	2H	48.4
36-27	MBM	1H	16.5
36-27	MBM	3H	5.7
36-28	MBM	5H	37.6
36-28	MBM	2C	22.7
36-29	MBM	1C	42.0
36-30	MBM	1C	42.0
36-32	MBM	1H	50.0
36-33	MBM	5H	15.5
36-44	MBM	4C	4.5
37-65	MBM	4H	5.8
38-26	MMB	2H	43.1
38-27	MBM	4H	5.9
39-26	MBM	3H	32.1
39-26	MBM	3H	35.6
39-26	MBM	3H	39.7
39-26	MBM	1H	44.6
39-26	MMB	3H	26.9
39-29	MBM	2H	35.6
39-29	MBM	3H	36.5
39-29	MBM	AV2	20.9
39-29	MBM	2H	33.8
39-41	MBM	2H	15.6
39-41	MBM	4H	45.0
39-53	MBM	6H	16.8

Row - Column	Indication	Location	Inch Mark
40-30	MBM	4H	12.6
40-37	MBM	1H	34.4
40-37	MBM	3H	14.0
40-59	MBM	1H	44.7
40-59	MBM	3H	33.3
40-59	MBM	3C	24.8
40-59	MBM	3C	34.9
41-38	MBM	1C	35.9
41-49	MBM	5C	25.3
42-39	MBM	3C	36.7
42-39	MBM	1H	8.5
42-53	MBM	4C	17.2
42-53	MBM	1H	39.9
43-37	MMB	2C	22.4
43-41	MBM	TSC	3.5
43-51	MBM	4H	44.8
45-43	23	AV1	0.0

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
BPH or C - Baffle Plate Hot or Cold Leg
DNG - Ding
MBM - Manufacturing Burnishing Marks

MMB - Multiple MBMs
NQN - Non-Quantifiable Indication - Not Reportable
TSH or C - Tube Sheet Hot or Cold Leg

Row - Column	Indication	Location	Inch Mark
2-68	MBM	4C	42.8
5-40	MBM	1C	16.2
5-78	MBM	2C	5.3
6-19	MBM	BPH	16.2
6-54	MBM	2H	4.9
6-59	MBM	1C	32.6
6-59	MBM	2C	26.4
6-63	MBM	TSH	11.7
6-63	MBM	TSH	8.5
7-22	MBM	1H	5.0
7-42	MBM	1C	42.9
7-63	MBM	TSC	22.3
8-7	MBM	1C	34.9
8-64	MBM	4C	15.4
9-40	MBM	TSH	46.3
9-82	MBM	1C	42.2
10-78	MBM	2C	30.5
12-14	MBM	3H	27.9
12-80	MBM	1H	46.4
12-82	MBM	2C	28.5
12-82	MBM	TSH	7.7
12-84	MBM	1C	12.3
12-84	MMB	2C	7.2
13-14	MBM	3H	32.9
13-79	MBM	4H	40.6

Row - Column	Indication	Location	Inch Mark
14-11	MBM	5H	39.4
14-72	MBM	4C	25.0
14-72	MBM	2H	16.9
14-78	MBM	4H	36.8
14-79	MBM	2C	33.3
14-87	MBM	3H	38.0
15-14	MBM	1C	44.3
15-76	MBM	5H	8.1
16-12	MBM	4H	25.5
16-12	MBM	4H	25.2
16-14	MBM	1H	9.1
16-74	MBM	2H	4.8
16-79	MBM	BPC	9.2
16-82	MBM	BPH	16.1
16-87	MBM	3C	39.3
17-16	MBM	4C	39.4
17-74	MBM	1H	10.3
17-79	MBM	2C	20.4
17-79	MBM	4H	11.0
17-82	MBM	3C	44.0
17-84	MBM	4H	5.3
17-85	MMB	BPH	5.8
18-14	MBM	1C	42.7
18-87	MBM	TSC	6.2
20-8	NQN	5C	24.1
21-10	MBM	5H	24.4
21-10	MBM	4H	22.2
21-13	MBM	4C	29.7
21-13	MBM	2H	31.3
21-18	MBM	2H	50.8
22-74	MBM	3H	42.5
22-74	MBM	5C	4.7

Row - Column	Indication	Location	Inch Mark
22-78	MBM	1H	13.5
22-81	MBM	BPH	4.6
22-81	MBM	BPH	6.1
22-84	MBM	1H	49.0
22-84	MBM	5H	26.1
23-19	MBM	1H	7.7
23-22	MBM	3C	42.4
25-22	MBM	5C	4.5
25-23	MBM	BPC	2.5
25-72	MBM	BPH	14.0
27-11	MMB	1C	1.0
28-71	MBM	TSH	9.2
30-14	MBM	3H	35.8
31-16	MBM	3C	19.8
31-16	MBM	BPC	17.2
31-16	MBM	4C	37.8
31-25	MBM	1H	10.3
31-26	MMB	TSH	15.4
31-27	MBM	5C	40.3
31-27	MBM	1H	2.7
31-27	MBM	2C	40.4
31-28	MBM	1H	29.4
31-28	MBM	1H	34.8
31-66	MBM	BPH	16.2
31-66	MBM	4H	7.6
32-14	MBM	BPC	23.4
32-65	MMB	TSH	9.9
32-65	MBM	BPH	13.1
32-66	MBM	1H	31.2
32-68	MBM	TSH	3.4
32-68	MMB	BPH	18.0
32-69	MBM	4H	29.3

Row - Column	Indication	Location	Inch Mark
32-70	MBM	3C	22.4
32-72	MBM	1H	50.9
33-20	MBM	3C	26.4
34-29	MBM	4H	30.4
34-29	MBM	4H	23.1
34-33	MBM	1H	8.3
34-33	MBM	BPC	23.4
34-33	MBM	4C	20.4
34-33	MBM	4H	44.0
34-76	MBM	BPC	20.9
35-51	16	AV1	-0.2
35-51	15	AV3	-0.0
35-51	30	AV2	0.1
35-75	MBM	BPH	16.7
37-33	MMB	2C	44.4
37-68	NQN	5H	17.6
37-68	DNG	5H	17.6
37-69	23	5H	12.3
37-70	MBM	2H	31.8
38-25	MBM	1C	16.6
38-25	MBM	5H	22.6
38-25	MBM	3H	24.0
38-26	MBM	4H	28.4
38-27	MBM	1C	30.6
38-27	MMB	4H	4.2
38-27	MBM	1C	4.5
38-29	MBM	5C	16.3
38-29	MMB	2H	16.0
38-30	MBM	1C	22.9
38-30	DNG	BPH	13.7
38-44	MBM	2C	27.7
38-49	MBM	5H	34.5

Row - Column	Indication	Location	Inch Mark
38-50	MBM	1C	50.0
38-50	MMB	2C	11.1
38-51	MBM	2H	46.4
38-51	MBM	2H	4.6
39-35	MBM	1H	41.7
39-37	MBM	5H	24.6
39-37	MBM	5H	19.3
39-48	MBM	3H	42.9
39-49	MBM	1H	47.3
39-49	MBM	BPH	21.1
39-51	MBM	4C	45.7
39-51	MBM	4C	21.1
39-52	MBM	4C	38.1
39-52	MBM	4C	25.3
39-52	MBM	3H	47.0
39-53	MBM	4H	7.9
39-56	MBM	1C	25.1
39-56	MBM	1C	17.0
39-56	MBM	1C	23.2
39-58	MBM	1H	7.8
39-58	MBM	BPH	9.8
39-59	MBM	TSH	3.5
39-59	MBM	TSH	1.4
39-59	MBM	TSH	14.5
39-59	MBM	1H	23.0
39-59	MBM	3C	23.1
39-59	MBM	3H	20.3
39-59	MBM	4H	23.8
39-59	MBM	4H	17.5
39-62	MBM	4C	31.6
39-62	MBM	1C	14.5
39-62	MBM	4H	12.7

Row - Column	Indication	Location	Inch Mark
39-62	MBM	3H	20.8
39-62	MBM	1H	27.5
39-62	MMB	1H	23.6
39-62	MBM	1H	3.2
39-64	MBM	TSH	6.0
39-64	MBM	BPC	19.1
39-64	MBM	BPC	15.8
39-64	MBM	BPC	8.1
39-64	MBM	2C	17.5
39-64	MBM	3C	46.6
39-64	MBM	3H	30.5
39-65	MBM	2C	12.1
39-65	MBM	1C	36.7
39-65	MBM	2C	9.7
39-65	MBM	5C	1.7
39-65	MBM	4H	47.9
39-65	MBM	TSH	15.9
39-65	MBM	1C	39.2
39-65	MBM	1C	50.5
39-66	MBM	5H	43.9
39-66	MBM	1C	14.0
39-66	MBM	2C	42.1
39-66	MMB	3C	13.8
39-66	MMB	3C	20.3
39-67	MBM	3H	22.1
39-67	MBM	4C	41.7
39-67	MBM	3H	11.6
39-67	MBM	3H	16.4
40-36	NQN	5H	24.2
40-36	DNG	5H	24.2
40-44	MBM	TSH	4.4
40-47	MBM	4H	33.0

Row - Column	Indication	Location	Inch Mark
40-66	MBM	TSH	6.2
41-49	MBM	2H	17.4
42-32	MBM	1C	8.8
43-49	MBM	4H	48.4
43-49	MBM	4H	42.0
43-49	MMB	3C	48.3
43-50	MBM	1H	44.0
43-50	MBM	5C	28.8
43-50	MBM	2C	15.7
43-50	MBM	2C	19.1
43-51	MBM	5H	18.9
43-51	MBM	1C	4.3
43-51	MBM	3C	46.1
43-51	MBM	2C	7.1
43-51	MMB	2C	15.4
43-51	MBM	1C	2.2
43-52	MMB	2H	6.2
43-56	MBM	1C	3.2
43-56	MBM	1H	13.6
43-58	MBM	BPC	11.1
44-36	MBM	BPH	2.3
44-39	MBM	BPH	24.1
44-40	MBM	BPC	4.8
45-41	DNG	BPH	2.6
45-41	DNG	BPH	2.6
45-41	DNG	BPH	2.8

UNIT 2

Inspection Plan: During the Unit 2 Refueling 19 outage, eddy current testing was performed October 9, 1993, to October 12, 1993. Full length eddy current testing was performed on 100% of the unsleeved tubes. In addition, 20% of the sleeved tubes were inspected and any cold leg unsleeved tube not included in any inspection plan, was inspected to the first support plate. 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	1416	1536
No. 1 TSP	(1121)	(535)
SLEEVES	301 (342)	280 (469)
RPC	46 (19)	34 (20)
Totals	1763 (1482)	1850 (1024)

RPC - Rotating write-up pancake coil inspections done to the extent necessary to bound distorted indications.

Inspection Results: The results of these inspections revealed 66 tubes in the "A" steam generator with reportable indications, and 104 in the "B" steam generator. The following is a summary of the eddy current inspection results listing the largest indication per tube:

Eddy Current Inspection Results Hot Leg (Cold Leg)		
	"A" SG	"B" SG
DI	10 (0)	2 (0)
DRI	8 (0)	0 (0)
20-29%	9 (24)	11 (45)
30-39%	2 (13)	11 (29)
40-49%	0 (1)	1 (5)
≥50%	1 (0)	1 (0)
NQI	2 (0)	3 (0)
Axial Ind	18 (0)	3 (0)
MBM	0 (0)	0 (0)
Totals	50 (38)	32 (79)

% - Percent Through Wall Indication
MBM - Manufacturing Burnishing Marks
DRI - Distorted Roll Indication
DI - Distorted Indication

NQI - Non-Quantifiable Indication

Repaired or Plugged Tubes: The following lists the tubes which were mechanically plugged during the Unit 2 Refueling 18 outage:

NOTE: All inch marks are above the referenced location unless otherwise specified.

Plugged Tube in the "A" Steam Generator			
Row - Column	Indication/ %	Location	Inch Mark
7-1	43%	1C	0.2
28-14	DI	TEH	5.4
	SAI	TEH	6.8
	DIM	TEH	0.0
2-19	DTI	TSH	0.3
	RST	UB	0.0
	SAI	TSH	0.2
	SAI	TSH	0.2
36-41	DRI	TEH	2.5
	MAI	TEH	2.5
	DIM	TEH	0.0
41-45	DRI	TEH	2.5
	SAI	TEH	7.7
	DIM	TEH	0.0
	MAI	TEH	4.3
	DIM	TEH	0.0
	SAI	TEH	2.8
	DIM	TEH	0.0
33-47	86%	TEH	17.1
	DTI	TSH	-0.3
	DI	TSH	-1.0
	SAI	TEH	14.7
	DIM	TEH	0.0
	SAI	TSH	-0.8
	DIM	TSH	0.0
34-48	DRI	TEH	2.6
	MAI	TEH	1.5

Plugged Tube in the "A" Steam Generator			
Row - Column	Indication/ %	Location	Inch Mark
	DIM	TEH	0.0
41-50	TEH	TEH	6.4
	17%	TSH	13.5
	DI	TEH	7.0
	12%	TSH	13.6
	MAI	TEH	7.7
	DIM	TEH	0.0
41-52	DRI	TEH	2.7
	MAI	TEH	2.7
	DIM	TEH	0.0
37-61	DRI	TEH	1.9
	SAI	TEH	1.8
	DIM	TEH	0.0
34-63	DI	TEH	4.2
	MAI	TEH	3.3
	DIM	TEH	0.0
37-69	DI	TEH	5.8
	MAI	TEH	6.3
	DIM	TEH	0.0
35-69	DI	TEH	6.4
	MAI	TEH	5.2
	DIM	TEH	0.0
35-71	NQI	TEH	5.5
30-72	DI	TEH	4.8
	SAI	TEH	3.2
	DIM	TEH	0.0
30-73	DI	TEH	4.0
	SAI	TEH	5.0
	DIM	TEH	0.0
4-74	NQI	TEH	4.7
34-76	DI	TEH	2.9

Plugged Tube in the "A" Steam Generator			
Row - Column	Indication/ %	Location	Inch Mark
	SAI	TEH	2.5
34-76	DIM	TEH	0.0
30-76	DRI	TEH	2.5
	DI	TEH	4.3
	DI	TEH	2.7
	SAI	TEH	5.7
	DIM	TEH	0.0
	MAI	TEH	4.7
	DIM	TEH	0.0
17-85	DRI	TEH	2.4
	MAI	TEH	2.0
	DIM	TEH	0.0
11-89	DRI	TEH	2.2
11-89	MAI	TEH	1.9
	DIM	TEH	0.0

% - Percent Through Wall
1H or 1C Indication - Support Plate Number Hot or Cold Leg
MAI - Multiple Axial Indication
SAI - Single Axial Indication
TEH - Tube End Hot Leg
TSH - Tube Sheet Hot Leg
DTI -Distorted Tubesheet Indication

DI - Distorted Indication
DRI - Distorted Roll
RST - Restricted
NQI - Non-Quantifiable Indication
DIM - Dimension

Plugged Tube in the "B" Steam Generator			
Row - Column	Indication/ %	Location	Inch Mark
3-1	40%	1C	0.0
11-4	NQI	TEH	5.6
	RST	1H	0.0
	RST	1H	0.0
13-4	43%	1C	0.1
15-13	NQI	TEH	6.1
22-14	DI	TEH	2.7
	SAI	TEH	4.2

Plugged Tube in the "B" Steam Generator			
Row - Column	Indication/ %	Location	Inch Mark
	DIM	TEH	0.0
31-16	DI	TEH	8.8
	DI	TEH	9.1
	SAI	TEH	9.1
	DIM	TEH	0.0
29-19	NQI	TEH	3.4
	18%	1C	0.2
1-45	DTI	TSH	1.2
	SAI	TSH	0.8
	DIM	TSH	0.0
26-51	48%	1C	-0.1
26-63	41%	AV3	0.0
20-68	41%	TSC	0.9
23-69	40%	1C	0.0

% - Percent Through Wall
1H or 1C - Support Plate Number Hot or Cold Leg
MAI - Multiple Axial Indication
SAI - Single Axial Indication
TEH - Tube End Hot Leg
DTI -Distorted Tubesheet Indication
NQI -Non-Quantifiable Indication
TSH - Tube Sheet Hot Leg
RST - Restricted
DI - Distorted Indication
DIM - Dimension

Tubes with Indications - Not Plugged: The following is a list of tubes with indications not repaired during Unit 2 Refueling 19 outage as a result of eddy current indications.

"A" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
1-12	DIN	TSC	16.9
1-18	28	1H	19.4
1-21	DTN	TSH	0.1

Row - Column	Indication	Location	Inch Mark
1-24	21	1H	21.3
1-24	24	1H	27.3
1-36	23	1H	21.4
1-91	33	1C	-0.2
2-19	SAI	TSH	0.2
2-19	DTI	TSH	0.3
2-19	SAI	TSH	0.2
2-39	11	TSC	7.9
2-88	DRN	TEH	2.3
2-89	DRN	TEH	2.0
2-90	DRN	TEH	2.2
3-18	DTN	TSH	0.4
3-19	DTN	TSH	0.0
3-77	DTN	TSH	0.2
4-16	DTN	TSH	0.0
4-19	DTN	TSH	0.1
4-31	22	TSC	1.2
4-34	11	TSC	1.6
4-74	NQI	TEH	4.7
4-76	DTN	TSH	0.3
4-77	DTN	TSH	0.3
5-16	DTN	TSH	0.0
5-31	37	TSC	0.5
5-31	23	TSC	0.9
5-35	13	TSC	1.4
5-71	DTN	TSC	0.4
5-74	13	TSC	2.0
6-25	27	TSC	0.6
6-26	9	TSC	1.3
6-26	19	TSC	9.9
6-33	23	TSC	0.6
6-34	23	TSC	0.3

Row - Column	Indication	Location	Inch Mark
6-35	13	TSC	0.6
6-37	DTN	TSC	0.5
6-38	DIN	TSC	0.7
6-50	DIN	TSC	5.2
6-81	DRN	TEH	1.9
6-86	DRN	TEH	1.8
7-1	DIM	1C	95.0
7-1	43	1C	0.2
7-23	14	TSC	1.7
7-30	30	TSC	0.7
7-33	18	TSC	0.4
7-34	25	TSC	0.6
7-35	15	TSC	1.8
7-35	16	TSC	1.0
7-37	37	TSC	0.8
7-37	22	TSC	1.1
7-78	29	TSH	0.4
8-24	31	TSC	1.3
8-34	36	TSC	0.5
8-36	37	TSC	0.7
8-36	21	TSC	1.4
8-36	22	TSC	2.1
8-46	18	TSC	1.7
9-5	DSN	1H	0.0
9-25	22	TSC	0.5
9-28	26	TSC	0.8
9-35	10	TSC	0.8
9-37	DTN	TSC	0.0
9-38	18	TSC	1.9
9-40	5	TSC	1.9
9-47	DIN	TSC	1.3
9-48	23	TSC	1.1

Row - Column	Indication	Location	Inch Mark
9-58	32	TSC	0.0
10-25	19	TSC	0.5
10-26	DTN	TSC	0.4
10-28	16	TSC	0.7
10-48	30	TSC	1.2
10-66	22	TSC	0.3
10-79	DTN	TSH	0.2
11-47	31	TSC	0.7
11-66	17	TSC	0.4
11-77	DTN	TSH	0.4
11-89	MAI	TEH	1.9
11-89	DIM	TEH	0.0
11-89	DRI	TEH	2.2
12-28	28	TSC	6.8
12-30	UDS	2C	24.7
12-32	8	TSC	0.7
12-64	DTN	TSC	0.3
12-64	DTN	TSC	0.5
13-25	16	TSC	0.9
13-33	10	TSC	0.7
13-37	15	TSC	1.2
13-47	32	TSC	1.0
13-73	DTN	TSC	0.0
13-88	23	TSC	14.3
14-26	DTN	TSC	0.4
14-31	DTN	TSC	0.5
14-32	18	TSC	0.8
14-39	17	TSC	0.5
14-40	12	TSC	0.6
14-48	23	TSC	1.3
14-48	21	TSC	0.7
15-31	13	TSC	0.6

Row - Column	Indication	Location	Inch Mark
15-39	14	TSC	0.5
16-5	DSN	2H	0.0
16-30	18	TSC	0.9
16-32	19	TSC	0.8
16-48	20	TSC	0.9
17-6	DSN	1H	0.0
17-20	12	TSC	16.7
17-24	DTN	TSC	0.0
17-38	18	TSC	0.6
17-47	11	TSC	0.6
17-85	DRI	TEH	2.4
17-85	MAI	TEH	2.0
17-85	DIM	TEH	0.0
18-5	25	1H	0.2
18-6	38	1H	0.1
18-47	24	TSC	0.4
18-48	16	TSC	0.8
18-59	12	TSC	22.4
18-59	16	TSC	21.2
18-60	26	TSC	11.2
19-30	29	TSC	1.0
19-30	21	TSC	9.5
19-78	DRN	TEH	1.7
19-79	DRN	TEH	1.7
19-83	DIN	1H	0.6
20-12	23	TSH	7.7
20-25	22	TSC	0.5
20-45	DIN	TSC	1.2
20-53	14	TSC	2.1
20-87	DRN	TEH	1.8
21-28	DTN	TSC	0.0
21-33	13	TSC	22.5

Row - Column	Indication	Location	Inch Mark
21-37	15	TSC	6.8
21-56	12	TSC	29.1
22-7	28	2H	0.1
22-36	23	TSC	0.8
23-82	10	TSC	31.2
25-8	DSN	1H	0.0
28-14	DI	TEH	5.4
28-14	DIM	TEH	0.0
28-14	SAI	TEH	6.8
30-72	DI	TEH	4.8
30-72	SAI	TEH	3.2
30-72	DIM	TEH	0.0
30-73	SAI	TEH	5.0
30-73	DIM	TEH	0.0
30-73	DI	TEH	4.0
30-76	DIM	TEH	0.0
30-76	DI	TEH	2.7
30-76	DI	TEH	4.3
30-76	DRI	TEH	2.5
30-76	DIM	TEH	0.0
30-76	MAI	TEH	4.7
30-76	SAI	TEH	5.7
33-16	DSN	1C	0.0
33-47	86	TEH	17.1
33-47	SAI	TEH	14.7
33-47	DIM	TEH	0.0
33-47	SAI	TSH	-0.8
33-47	DIM	TSH	0.0
33-47	DTI	TSH	-0.3
33-47	DI	TSH	-1.0
34-48	DIM	TEH	0.0
34-48	MAI	TEH	1.5

Row - Column	Indication	Location	Inch Mark
34-48	DRI	TEH	2.6
34-63	MAI	TEH	3.3
34-63	DI	TEH	4.2
34-63	DIM	TEH	0.0
34-65	17	TSC	44.5
34-65	13	TSC	50.4
34-65	11	TSC	49.3
34-75	DIN	TEH	16.9
34-76	SAI	TEH	2.5
34-76	DI	TEH	2.9
34-76	DIM	TEH	0.0
35-69	DIM	TEH	0.0
35-69	DI	TEH	6.4
35-69	MAI	TEH	5.2
35-71	NQI	TEH	5.5
36-41	MAI	TEH	2.5
36-41	DIM	TEH	0.0
36-41	DRI	TEH	2.5
36-64	11	TSC	46.5
36-64	26	TSC	43.9
37-61	DIM	TEH	0.0
37-61	SAI	TEH	1.8
37-61	DRI	TEH	1.9
37-69	MAI	TEH	6.3
37-69	DIM	TEH	0.0
37-69	DI	TEH	5.8
38-50	22	TSC	28.5
38-50	10	TSC	28.2
38-50	30	TSC	26.9
38-66	10	TSC	35.0
39-50	22	1H	14.9
39-50	15	1H	10.9

Row - Column	Indication	Location	Inch Mark
39-51	DSN	1H	0.2
39-59	35	TSC	23.7
40-26	36	TSH	5.4
40-48	10	TSC	41.7
40-48	11	TSC	32.0
40-48	18	TSC	34.7
40-61	15	6H	27.0
41-45	SAI	TEH	7.7
41-45	DIM	TEH	0.0
41-45	DRI	TEH	2.5
41-45	SAI	TEH	2.8
41-45	DIM	TEH	0.0
41-45	MAI	TEH	4.3
41-45	DIM	TEH	0.0
41-48	10	TSC	42.0
41-48	12	TSC	37.6
41-48	10	TSC	42.2
41-48	25	TSC	33.1
41-50	DI	TEH	6.4
41-50	DIM	TEH	0.0
41-50	17	TSH	13.5
41-50	DI	TEH	7.0
41-50	12	TSH	13.6
41-50	MAI	TEH	7.7
41-51	DTN	TSH	0.5
41-52	DIM	TEH	0.0
41-52	DRI	TEH	2.7
41-52	MAI	TEH	2.7
41-61	19	TSC	35.3
41-65	11	TSC	42.2
42-51	27	TSH	0.4
42-53	15	TSH	15.6

Row - Column	Indication	Location	Inch Mark
43-44	16	1H	0.0
44-52	14	TSH	2.6
45-52	19	2H	0.0

"B" Steam Generator Indications

1-6H or C - Tube Support Plate Number Hot or Cold Leg
AV1-4 - Anti-Vibration Bar Number
TEH or C - Baffle Plate Hot or Cold Leg
TSH or C - Tube Sheet Hot or Cold Leg
DIM - Dimension
DIN - Distorted Indication - Not Reportable
DRN - Distorted Roll Indication - Not Reportable
DTI - Distorted Tubesheet Indication
DTN - Distorted Tubesheet Indication - Not Reportable
DSN - Distorted Support Indication - Not Reportable
NQI - Non-Quantifiable Indication
SAI - Single Axial Indication

NOTE: All inch marks are above the referenced location unless otherwise specified.

Row - Column	Indication	Location	Inch Mark
1-2	31	1C	0.0
1-21	9	1C	0.0
1-44	DTN	TSH	1.3
1-45	DIM	TSH	0.0
1-45	SAI	TSH	0.8
1-45	DTI	TSH	1.2
1-62	DTN	TSH	0.6
1-65	25	1H	-0.2
2-1	22	TSC	8.8
2-15	11	1H	17.8
2-15	25	1H	18.0
2-17	DTN	TSH	0.4
2-50	DIN	TSC	0.6
2-59	12	TSC	1.6
2-80	DRN	TEH	2.2
2-83	DRN	TEH	1.1
3-1	40	1C	0.0

Row - Column	Indication	Location	Inch Mark
3-19	DTN	TSH	0.3
3-55	13	TSC	0.4
3-67	32	TSC	1.3
3-69	13	TSC	0.8
3-71	DRN	TEH	2.3
4-17	DTN	TSH	0.3
4-18	DTN	TSH	0.2
4-19	DTN	TSH	0.4
4-28	DIN	TSC	0.3
4-35	14	TSC	0.3
4-71	29	TSC	0.4
5-2	21	1C	-0.1
5-4	18	1H	32.3
5-50	15	TSC	0.3
6-14	DTN	TSH	0.6
6-15	36	TSH	39.8
6-16	DTN	TSH	0.3
7-1	12	1H	43.6
7-15	21	1H	0.1
7-72	DTN	TSC	0.0
7-81	DRN	TEH	2.3
8-17	DTN	TSH	0.3
8-73	DTN	TSC	0.1
9-73	DTN	TSC	0.1
9-74	DIN	TSC	1.0
9-81	30	1H	19.1
10-74	DIN	TSC	1.1
10-76	DIM	TSH	0.0
10-76	11	TSH	0.2
11-4	NQI	TEH	5.6
11-29	21	TSC	1.1
11-73	DTN	TSC	0.2

Row - Column	Indication	Location	Inch Mark
11-73	12	TSC	1.1
11-88	18	1C	0.0
11-91	DSN	1H	0.0
12-71	13	TSC	0.9
12-72	15	TSC	1.1
12-73	DTN	TSC	0.2
13-4	43	1C	0.1
13-71	22	TSC	1.0
13-90	DIN	TSH	1.3
14-37	13	TSC	0.8
14-72	13	TSC	0.7
14-89	32	3C	0.0
15-13	NQI	TEH	6.1
15-33	11	TSC	1.1
15-62	13	TSC	0.9
17-5	17	1H	5.0
17-21	35	6H	0.0
17-22	17	1C	-0.1
17-36	26	TSC	0.3
19-31	DIN	TSC	0.5
19-37	25	TSC	0.8
19-70	DTN	TSC	0.0
19-77	36	AV1	0.0
20-58	29	TSC	0.9
20-68	41	TSC	0.9
20-69	DTN	TSC	0.0
21-7	29	1C	0.0
21-31	22	TSC	0.8
21-60	16	TSC	0.7
22-7	22	TSH	40.5
22-7	27	TSH	38.6
22-7	18	TSH	39.6

Row - Column	Indication	Location	Inch Mark
22-7	29	TSH	38.6
22-14	DI	TEH	2.7
22-14	DIM	TEH	0.0
22-14	SAI	TEH	4.2
22-19	27	TSH	30.7
22-24	35	1C	0.0
22-28	23	1C	-0.2
22-35	UDS	6H	48.7
22-86	37	1C	-0.2
23-19	DIN	TEH	5.4
23-29	32	1C	-0.2
23-42	26	TSC	0.4
23-69	40	1C	0.0
24-32	22	TSC	2.5
24-44	18	TSC	0.6
25-46	17	TSC	0.9
25-64	15	TSC	0.6
25-77	DRN	TEH	2.2
26-12	12	1C	0.0
26-13	31	1C	0.0
26-24	27	1C	-0.1
26-47	DIN	TSC	0.6
26-51	48	1C	-0.1
26-62	13	TSC	5.7
26-63	41	AV3	0.0
26-68	34	TSC	1.4
26-73	14	1H	18.5
26-75	18	1H	14.1
27-29	22	1C	0.0
27-69	31	1C	0.0
27-69	25	1C	0.1
27-79	25	TSH	3.8

Row - Column	Indication	Location	Inch Mark
27-79	15	TSH	6.3
28-16	33	TSH	43.3
28-16	29	TSH	47.3
28-23	DTN	TSH	0.4
28-24	DTN	TSH	0.2
28-35	11	4H	-0.2
28-39	23	1C	0.0
28-49	16	TSC	2.2
29-16	34	1C	0.1
29-19	18	1C	0.2
29-19	NQI	TEH	3.4
29-26	33	1C	0.0
29-28	20	1C	-0.1
29-30	13	1C	0.0
29-30	13	1H	11.3
29-41	27	1C	0.0
29-66	DRN	TEH	2.4
29-70	DRN	TEC	1.9
30-19	DIN	TEH	7.6
30-36	37	1C	0.0
30-41	18	1C	0.0
30-47	22	1C	0.0
30-48	20	1C	0.0
30-49	31	1C	0.0
30-70	DRN	TEC	2.0
30-75	DIN	TSC	3.8
31-16	DI	TEH	8.8
31-16	DIM	TEH	0.0
31-16	SAI	TEH	9.1
31-16	DI	TEH	9.1
31-28	22	1C	-0.1
31-36	32	1C	0.0

Row - Column	Indication	Location	Inch Mark
31-66	39	1C	0.1
31-67	32	1C	0.0
31-70	DRN	TEC	1.9
31-71	27	1H	8.6
32-20	24	1C	0.0
32-21	18	1C	-0.1
32-35	22	1H	18.4
32-38	21	1C	0.0
32-70	37	1C	0.0
32-70	22	1C	-0.1
33-19	25	1C	0.2
33-20	18	1C	0.1
33-46	35	1C	0.0
33-48	34	1C	0.0
33-48	DTN	TSH	0.8
33-51	13	1C	0.1
33-58	27	1C	0.0
33-60	29	1C	0.1
33-73	34	1C	-0.1
34-22	13	1C	0.0
34-23	10	1C	0.0
34-39	30	1C	0.0
34-39	30	1H	7.8
34-45	DTN	TSH	0.1
34-52	22	1C	0.0
34-53	13	1C	0.0
34-58	DRN	TEH	1.5
34-67	37	1C	0.0
34-67	DRN	TEH	2.2
34-72	DRN	TEH	2.3
35-35	18	1H	20.2
35-58	DRN	TEC	2.0

Row - Column	Indication	Location	Inch Mark
36-21	19	1C	0.0
36-22	14	1C	0.0
36-25	24	1C	0.0
36-45	DTN	TSH	0.6
36-48	11	TSH	35.9
36-48	19	TSH	42.8
36-63	25	1C	0.0
36-65	22	1C	0.1
36-66	36	1C	0.0
37-21	24	1C	0.0
37-23	22	1C	0.0
37-27	27	1C	0.0
37-48	12	1C	0.0
37-50	11	1C	0.0
37-61	28	1C	0.0
37-62	33	1C	0.1
37-63	22	1C	0.2
37-67	18	1C	0.1
37-68	25	1C	0.0
38-25	35	TSH	50.9
38-33	14	1H	28.8
38-48	29	1C	-0.1
38-48	16	1C	0.1
34-49	22	1C	-0.1
38-52	32	1C	0.0
38-54	24	1C	0.0
38-55	13	1C	0.0
38-61	32	AV4	0.0
39-24	22	1C	0.0
39-25	33	1C	0.0
39-27	25	1C	0.0
39-35	35	1C	0.0

Row - Column	Indication	Location	Inch Mark
39-53	DTN	TSH	0.0
39-65	10	1C	0.0
40-57	DSN	1C	0.2
40-66	DRN	TEH	2.4
41-29	18	TSH	23.2
41-46	33	1C	0.0
41-47	26	1C	-0.1
41-48	26	1C	0.0
42-41	14	TSH	35.5
42-49	31	1C	0.0
42-60	13	TSH	0.8
42-60	28	TSH	6.4
43-58	34	TSH	2.2
43-60	38	TSH	2.6
43-60	23	TSH	2.9
45-47	39	2H	0.2

VII. REACTOR COOLANT SYSTEM RELIEF VALVE CHALLENGES

Overpressure Protection During Normal Pressure and Temperature Operation

There were no challenges to the Unit 1 or Unit 2 reactor coolant system power-operated relief valve or safety valves at normal operating pressure and temperature in 1993.

Overpressure Protection During Low Pressure and Temperature Operation

There were no challenges to Unit 1 or 2 power-operated relief valves during low temperature and low pressure operation in 1993.

VIII. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications during operation of Unit 1 or Unit 2 in 1993 where reactor coolant activity exceeded that allowed by Technical Specifications.