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NPL-96-0( 5 10 CFR 50.59

February 28, 1996

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Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
ANNUAL RESULTS AND DATA REPORT - 1995
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed are ten copies of the 1995 Annual Results and Data Report for Point Beach Nuclear Plant, Units 1 and 2. This report is submitted in accordance with Technical Specification 15.6.9.1.B pursuant to the requirements of 10 CFR 50.59(b). The report contains information regarding highlights of Point Beach Nuclear Plant operations during 199: and includes descriptions of facility changes, tests and experiments, personnel occupational exposures, results of steam generator in-service inspections, and listings of reactor coolant system relief valve challenges.

Sincerely,

Gary M. Krieser
Manager
Industry & Regulatory Services

Enclosures

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

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POINT BEACH NUCLEAR PLANT

UNITS 1 AND 2

ANNUAL RESULTS AND

DATA REPORT

1995

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

# PREFACE

This Annual Results & Data Report for 1995 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, 1995, through December 31, 1995, included a 37-day refueling/maintenance outage. Major work items included: steam generator eddy current testing; residual heat removal rotating assembly work; reactor vessel head shielding; fuel transfer tube shielding; G-03 emergency diesel generator Phase 3B tie-in work; control rod drive mechanism current order timing change modification; degraded grid voltage relay work; steam generator blowdown heat exchanger replacement; crossover steam dump pressure switches replacement; 1A-05 bus 2/3 loss of voltage logic work; component cooling small bore pipe seismic upgrades; SC-955 containment isolation valve retubing; 1RC-558 valve replacement; and, Bulletin 79-14 upgrades to service water and component cooling supports. In February the unit was reduced to 293 Mwe because of a rod that inadvertently dropped. The stationary gripper fuses were replaced. In July a turbine/reactor trip occurred because of electrical generator problems related to hot/humid weather conditions coupled with low lake water temperatures. Also in July, the unit was reduced to approximately 55% power to repair the 1P-28B main feedwater pump.

Unit 1 operated at an average capacity factor of 89.3% (MDC net) and an electrical/thermal efficiency of 34%. The unit and reactor availability were 88.7% and 89.2%, respectively. Unit 1 generated its 83 billionth kilowatt hour on January 6, 1995; its 84 billionth kilowatt hour on May 6, 1995; its 85 billionth kilowatt hour on July 30, 1995; and its 86 billionth kilowatt hour on October 20, 1995.

## UNIT 2

Highlights for the period January 1, 1995, through December 31, 1995, included a 57 day refueling/maintenance outage. Major work items included: steam generator eddy current testing and Plus-Point inspections and re-roll repairs; boric acid and reactor makeup flow transmitter replacements; rod insertion limit computer replacements; fuel transfer tube shielding; Phase 3C of new emergency diesel generator tie-in work; 2MS-2016 atmospheric steam dump valve replacement; HP turbine exhaust crossunder piping replacement; 480 V station transformers anchorage modifications; degraded grid voltage relays-bypass time delay relays; rod control cabinet anchorage of 2C41; SC-955 containment isolation valve retubing; 2A-05 bus 2/3 loss of voltage logic; main steam condensate dump valve replacements; addition of test points in Unit 2 safeguards circuits; and, ALARA shielding for reactor coolant pump cubicles. In February the unit was maintained in hot shutdown because of seal oil inleakage into the Unit 2 generator. In March the reactor tripped, because of a leak in the main turbine EH control oil system that shut a main steam stop valve. In May the unit was reduced to 55% power to support repairs of 2P-28B, the main feedwater pump.

Unit 2 operated at an average capacity factor of 79.7% (MDC net) and an electrical/thermal efficiency of 33.7%. The unit and reactor availability were 81.5% and 83%, respectively. Unit 2 generated its 83 billionth kilowatt hour on January 4, 1995; its 84 billionth kilowatt hour on April 6, 1995; its 85 billionth kilowatt hour on June 29, 1995; and its 86 billionth kilowatt hour on September 20, 1995.

## III. AMENDMENTS TO FACILITY OPERATING LICENSES

During 1995 there were 8 amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1 and 8 amendments issued to Facility Operating License DPR-27 for Point Beach Nuclear Pant Unit 2. The license amendments are listed by date of issue and summarized below.

Amendment 160 to DPR-24, Amendment 164 to DPR-27, January 18, 1995: The amendments change the operating conditions and limiting conditions for operation of containment and containment isolation systems and components.

Amendment 161 to DPR-24, Amendment 165 to DPR-27, March 6, 1995: The amendments revise TS 15.3.3; 15.3.4; 15.3.5; 15.3.7; 15.3.14; 15.4.1 to delete obsolete TS. It provides Spring, 1995 outage-specific TS as part of the ongoing diesel upgrade project and updates several TS to be consistent with the upgrade project design changes. A monthly testing requirement was also changed.

Amendment 162 to DPR-24, Amendment 166 to DPR-27, July 5, 1995: The amendments modify TS 15.6.5, "Review and Audit," and TS 15.7.8, "Administrative Controls," to relocate audit frequencies and emergency plan reviews to the appropriate controlling documents.

Amendment 163 to DPR-24, Amendment 167 to DPR-27, October 12, 1995: The amendments modified the General Considerations Section TS 15.3.0 to be consistent with Standard Technical Specification conditions including actions to be taken for situations not directly addressed in the action statements of TS.

Amendment 164 to DPR-24, Amendment 168 to DPR-27, October 12, 1995: The amendments removed the requirement for the Operations Manager to hold a SRO license to provide added staffing flexibility.

Amendment 165 to DPR-24, Amendment 169 to DPR-27, November 17, 1995: The amendments revise TS to reduce the reactor coolant system raw measured total flow rate limit and reflect the new reactor core safety limits for Unit 2.

Amendment 166 to DPR-24, Amendment 170 to DPR-27, November 22, 1995: The amendments implemented F* repair criteria for steam generator tubes in Unit 2 experiencing degradation in the portion of the tubes within the tubesheet region.

Amendment 167 to DPR-24, Amendment 171 to DPR-27, December 27, 1995: The amendments revise TS Tables 15.3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits," and 15.3.5-3, "Engineered Safety Features," to modify setting limits for degraded voltage protection and correct other references.

## IV. 10 CFR 50.59 & 10 CFR 72.48 SAFETY EVALUATIONS

## PROCEDURE CHANGES

The following procedure changes were implemented as of the end of 1995:

1. B&W42-EC-256, Multi-frequency Eddy Current Procedure, Steam Generator Tubing Digital Eddy Current System, Point Beach, Revision 0. (New Procedure)

   B&W 1154835A, Field Procedure for Remote and Manual Rolled Plugging, Revision 35. (New Procedure)

   The procedures govern the conduct of steam generator (SG) eddy current testing and tube plugging.

   Summary of Safety Evaluation: TS 15.3.1.G.3 list the operational limit on RCS flow that must be maintained for rated power operation. Performance of SG tube plugging (in which one SG tube is plugged) has a negligible effect on RCS flow. As the TS limit is approached, however, the overall percentage of tubes plugged must be monitored. Eddy current testing and tube plugging pose no threat to the integrity of the primary pressure boundary.

   FSAR Chapter 14 describes the accidents where the steam generators are utilized for decay heat removal. As long as RCS flow is greater than the TS limit, tube plugging does not effect the core damage frequency or fission product release frequency, as modeled by probabilistic safety assessment (PSA) calculations. FSAR Chapter 14.2.4 describes a steam generator tube rupture (SGTR) accident. Steam generator eddy current testing and tube plugging may decrease the probability of a SGTR because degraded tubes are removed from service. A minuscule amount of material may be removed from the tube end while cleaning the tube before plugging, but the final condition of the tube provides a greater degree of integrity. Proper foreign material exclusion procedures are utilized. (SER 95-041)

2. B&W 1191801A, Recirculating Steam Generator Mechanical Plug Retainers (PAP) Installation, Revision 13. (Permanent)

   B&W 1198350A, 7/8" Ribbed Plug Alloy 690 Retainer Development Report 51-1178025,00; Stress Report 33-1178999-00 for 7/8" Ribbed Plug (Alloy 690) Retainer; Equipment Specification 08-1202248-00 for W-F Ribbed Plug Retainer; and SG Tubesheet Machining Field Procedure, Revision 10. (Permanent)

   B&W 02-1210920A, Process Traveler Remote Welded Plug Installation Report 50-1212515-04; Weld Procedure Specification for 7/8" Remote Welded Plug Report 51-1205344-02; Stress Report 33-1205347-00 for 0.875 Remote Welded Plug; Equipment Specification 08-1178511-00 for Welded Tube Plugging of Heat Exchanger, and VT-1 Visual Examination of SG Tube Plug Welds, Revision 2. (Permanent)

The procedures relate to steam generator tube plug repairs. Repair of a SG tube may involve installation of a plug retainer (PAP), or drilling out the old plug and replacing it with a new rolled plug or a new welded plug. Choice of a specific repair method depends on location of the tube, whether the tube has an installed sleeve, amount of leakage exhibited by the tube, and condition of the tube end after machining or drilling is complete.

Summary of Safety Evaluation: The plug retainers and new plugs are designed to meet design criteria contained in FSAR Chapter 4, "Reactor Coolant System." They are designed for a lifetime of 40 years, and are compatible with primary and secondary water chemistry. They are designed to remain in place during normal operating and accident conditions. Performance of tube repairs does not adversely affect the pressure retaining function or the structural integrity of the tube or tubesheet as only a minuscule amount of tube material is removed during machining. This is done in order to effect a better quality weldment. Precautions are taken to ensure no foreign material remains in the steam generator.

Steam generator tube plug repairs are performed while the core is defueled. Therefore, no containment integrity or shutdown risk concerns exist. (SER 95-107)

3. Bechtel POP-1-G-03-02, Preoperational Test: G-03 Interface With 2A-06, Revision 0. (New Procedure).

The test verifies G-03 output breaker (2A52-87) to 4160V bus 2A-06 functions properly while the breaker is in the "test" position. The test also verifies that G-03 emergency diesel generator (EDG) starts in response to a simulated 2A-06 undervoltage signal.

Summary of Safety Evaluation: While the 2A52-87 breaker is aligned, it may receive an automatic signal in response to actual plant conditions (e.g., it may close on an actual 2A-06 undervoltage). With 2A52-87 closed in the "test" position, the G-04 output breaker to 2A-06 (breaker 2A52-93) does not automatically close. This is a design interlock that prevents G-03 and G-04 from simultaneously paralleling to the same bus. Therefore, G-04 is not operable to 2A-06 during this test. This requires entry into TS 15.3.7.B.1.f.a, a 7-day LCO for both units. The procedure contains a contingency to open 2A52-87 to allow 2A52-93 to automatically close for required plant conditions. The 2A52-93 breaker control circuits and interlocks were previously tested.

Temporary jumpers and lifted leads are installed in the 2A-06 cubicle. The jumpers and lifted leads involve contacts which are isolated from operating 2A-06 circuits. Terminals are independently verified prior to installation and after restoration. G-04 will start on a 2A-06 undervoltage or a Unit 2 SI signal and will remain available for accident mitigation. This test does not increase the probability of a malfunction.

G-04 EDG is load tested to 2A-06 in accordance with TS-84. This ensures that the 2A52-93 close circuit is restored prior to exiting the LCO. (SER 95-002)

4. CSP-C.1, Unit 1 and Unit 2, Response to Inadequate Core Cooling, Revision 11. (Permanent)

The evaluation addresses starting a RCP as a last effort to help provide core cooling and protect against creep rupture failure of steam generator (SG) tubes.

Summary of Safety Evaluation: Direct work item 93-019 changed the Westinghouse Owner's Group generic procedure to include a definition of an available reactor cooling loop as having SG level greater than [28%] 8%. This item is also being reviewed by the severe accident management working group that applied the EPRI Technical Basis Report (TBR) on creep rupture failure of SG U-tubes. The change ensures a RCP is only run when both SG levels are greater than [200"] 110" to prevent creep rupture failure of SG U-tubes. Wide range level on-scale is used instead of the WOG narrow range on-scale because SG level wide range is environmentally qualified and the generic plant is not.

The revision improves prevention of SG tube creep failure during a beyond design basis event by providing additional criteria prior to starting a RCP. If this procedure is entered, multiple failures have occurred to cause a loss of all core injections and core exit temperatures are above 1200°F. The change provides additional criteria for having both SG levels of on-scale to prevent creep rupture failure of the SG U-tubes. It also helps maintain the primary boundary. (SER 95-095)

5. EOP-1.3, Unit 1 and Unit 2, Transfer to Containment Sump Recirculation, Revision 13. (Temporary)

The change provided contingency actions to reduce the reliance on residual heat removal (RHR) from refueling water storage tank (RWST) suction check valves SI-854A&B as a reliable pressure boundary and to reestablish additional barriers (SI-850A&B, SI-851A&B, SI-856A&B) when possible. Steps were added to establish containment spray in the event a dynamic transfer to containment sump recirculation with a single train is required. Steps were also added to ensure a minimum RWST level is injected to address recriticality issue for U1C23 addressed in SER 95-050.

Summary of Safety Evaluation: Reestablishing additional barriers to the release of radioactivity reduces the consequences of leakage past a potentially degraded seat of RHR suction check valves SI-854A&B.

Starting a containment spray pump when performing a dynamic shift to containment sump RHR recirculation with a single SI train reduces the consequences of the potential SI-854A&B leakage by returning sump leakage past the suction check valves back to containment. If the valve seat is degraded, leakage is only predicted for a short period of time (approximately 4 minutes) while the containment sump valve is being opened, (SI-850) and the RWST suction valve is being shut (SI-856). After the containment spray pumps have been stopped, remaining sump water that may be present in the suction line to the RWST could be released from the system and not result in exceeding 10 CFR 100 limits. (SER 95-053)

6. EOP-3, Unit 1, Steam Generator Tube Rupture, Revision 17. (Permanent)

EOP-3, Unit 2, Steam Generator Tube Rupture, Revision 18. (Permanent)

The revisions incorporate provisions during cooldown following a steam generator tube rupture accident which consider an intact steam generator. This is verified by the steam generator non-return valve shut in the event neither the ruptured steam generator nor intact steam generator main steam isolation valve shuts. A pressurizer level is changed from 14% to 10% and incorporates a change in spray additive tank level setpoint of 12% for containment spray termination criteria.

Summary of Safety Evaluation: Taking credit for the non-return valve prevents or delays transition to ECA-3.1, "SGTR with Loss of Reactor Coolant-Subcooled Recovery Desired." ECA-3.1 reduces SI flow instead of using SI termination criteria used in EOP-3 and may result in overfilling the ruptured steam generator. By staying in EOP-3 or delaying entry into ECA-3.1 the chance for a radiological release is reduced. (SER 95-021)

7. 1ICP-02.017; 1ICP-02.017-1; 2ICP-02.017; 2ICP-02.017-1, Reactor Protection System Trip Logic Post Refueling Test, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. The upgrade included a complete technical review, including Technical Specifications, FSAR, Master Data Book, drawings, vendor information, and a walk-down of accessible components.

Summary of Safety Evaluation: The test is performed prior to startup following refueling. It checks reactor protection system trip matrixes for Trains A and B including annunciation and status lights. Steps were added listing conditions necessary to verify equipment has been returned to service. Acceptance criteria statements describe conditions and results necessary to satisfy TS requirements.

The activities are performed while the reactor is shut down and supports portions of TS 15.4.1, "Reactor Protective Instrumentation." The procedures do not reduce the prescribed frequency requirements of TS Table 15.4.1-1 Items 1, 4, 9, and 11 and TS Table 15.4.1-2 Item 27. The test methodology does not differ from that used during previous surveillances. (SER 95-071)

8. 1ICP-02.020; 1ICP-02.020-1; 1ICP-02.020-2; 1ICP-02.020RD-1; 1ICP-02.020WH-1; 1ICP-2.020BL-1; 1ICP-02.020YL-1, Post-Refueling, Pre-Startup RPS and ESF Analog Surveillance Tests, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. The procedures were written in accordance with the Style Guide for Procedures and the Writers Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: The procedures are unit and channel specific. One procedure tests annunciators after reaching normal temperature and pressure. One procedure test checks that equipment has been returned to normal. These activities are required by TS and are described in the FSAR. The procedures are performed while the unit is shut down. The methodology is unchanged from previous practice. Recent engineering setpoint changes for ΔTsp1 and 2 and loop current corrections for setpoints for high steam flow and high-high steam flow bistables were included. (SER 95-037)

9. 1ICP-04.002; 1ICP-04.002-1; 1ICP-04.002.2; 1ICP-04.002.3; 2ICP-04.002; 2ICP-04.002-1; 1ICP-04.002-2; 2ICP-04.002-3, Flow and Level Outage Calibrations, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. They were written in accordance with the Style Guide for procedures and the Writer's Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: The procedures were made unit specific. Calibration of reactor flow transmitters is required by TS and is described in the FSAR. Calibration of pressurizer cold level, resistance temperature detector (RTD) manifold flow and containment spray pump discharge flow instruments are not required by TS but are included in this series of outage conditions necessary for performance and test equipment requirements. Prerequisite conditions, TS frequency requirements, and acceptance criteria sections were added. The test methodology did not change from previous practice. (SER 95-023)

10. 1ICP-04.003; 1ICP-04.003-1; 1ICP-04.003-2; 1ICP-04.003-3; 1ICP-04.003-4; 1ICP-04.003-5; 1ICP-04.003-6; 1ICP-04.003-7; 1ICP-04.003-8; 1ICP-04.003-9; 2ICP-04.003; 2ICP-04.003-1; 2ICP-04.003-2; 2ICP-04.003-3; 2ICP-04.003-3; 2ICP-04.003-4; 2ICP-04.003-5; 2ICP-04.003-6; 2ICP-04.003-7; 2ICP-04.003-8; 2ICP-04.003-9, Level and Flow Instruments Outage Calibrations, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. They were written in accordance with the Style Guide for Procedures and the Writer's Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: The procedures are unit specific. Requirements were included for reactor vessel level to be at a stable condition prior to calibrating the level loops used for monitoring reactor vessel level during shutdown. TS frequency requirements and acceptance criteria sections were added. Attachments were included to list affected Control Room indications during calibration. This activity is performed during cold or refueling shutdown. Test methodology was not changed from previous practice. (SER 95-024)

11. 1ICP-04.004; 1ICP-04.004-1; 1ICP-04.004-2; 1ICP-04.004-3; 1ICP-04.004-4; 1ICP-04.004-5; 1ICP-04.004-6; 2ICP-04.004; 2ICP-04.004-1; 2ICP-04.004-2; 2ICP-04.004-3; 2ICP-04.004-4; 2ICP-04.004-5; 2ICP-04.004-6, Pressure Transmitter and Indicatoı Outage Calibrations, Revision 0. (New Procedures)

1ICP-04.006; 1ICP-04.006-1; 1ICP-04.006-2; 1ICP-04.006-3; 2ICP-04.006; 2ICP-04.006-1; 2ICP-04.006-2; 2ICP-04.006-3, Flow and Pressure Transmitters Outage Calibrations, Revision 0. (New Procedures)

1ICP-04.007; 1ICP-04.007-1; 1ICP-04.007-2; 2ICP-04.007; 2ICP-04.007-1; 2ICP-04.002-2, Feedwater Flow and SG Wide Range Level Outage Calibrations, Revision 0. (New Procedures)

1ICP-04.010; 1ICP-04.010-1; 1ICP-04.010-2; 2ICP-04.010; 2ICP-04.010-1; 2ICP-04.010-2, First Stage Pressure Transmitters and Auxiliary Feedwater Pressure Indication Outage Calibrations, Revision 0. (New Procedures)

1ICP-04.023; 1ICP-04.023-1; 2ICP-04.023; 2ICP-04.023-1, Reactor Vessel Level Outage Calibrations, Revision 0. (New Procedures)

The procedures were upgraded procedures based on existing instrumentation and control procedures. They were written in accordance with the Style Guide for Procedures and the Writer's Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: Procedures are unit specific. Prerequisite conditions, TS frequency requirements, and acceptance criteria sections were added. Attachments were included to list affected Control Room indications during calibration. These activities are performed during cold or refueling shutdown. Test methodology was not changed from previous practices. (SERs 95-025, 95-026, 95-027, 95-028, 95-029)

12. 1ICP-04.028; 1ICP-04.028-1; 2ICP-04.028; 2ICP-04.028-1, Independent Overspeed Protection System Outage Calibrations, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. New procedures are written in accordance with the Style Guide for Procedures and the Writer's Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: While this surveillance test is not described in the FSAR, it is required per TS Table 15.4.1-1 Item 42. Turbine generator overspeed is analyzed in FSAR Section 14.1.12. This test affects the operating unit; however, it is performed on a shutdown unit. The activity opens sliders in both units making both units of independent overspeed protection system (IOPS) inoperable while the test is being performed. TS 15.3.4.F requires only one turbine overspeed protection system that trips the turbine stop valves or shuts the turbine governor valves to be operable for the operating unit. Prerequisites were included to ensure this. The test does not disable the mechanical or auxiliary governor overspeed trips on the operating unit.

The IOPS input to the crossover steam dump system is made inoperable when the sliders are opened. The shutdown unit crossover steam dump system is made inoperable by opening two 125 Vdc breakers for crossover steam dump valves A and B. The at-power unit crossover steam dump valves are still operable but electrohydraulic control (EHC) contacts originating from a single overspeed protection controller card make the crossover steam dump system susceptible to a single failure mode. The crossover steam dump system was not designed as single failure proof. Therefore, the crossover steam dump system does not need two diverse redundant inputs. As a conservative measure, an administrative limit is placed on the time that the IOPS system can be out-of-service. Either or both units may be out-of-service for up to 72 hours. This activity has no effect upon accident probability. The test methodology does not vary from previous practice. (SER 95-038)

13. 1ICP-04.032; 1ICP-04.032-1; 2ICP-04.032; 2ICP-04.032-1, Auxiliary Feedwater System and Charging Flow Electronic Outage Calibrations, Revision 0. (New Procedures)

The procedures were upgraded based on existing instrumentation and control procedures. The new procedures were written in accordance with the Style Guide for Procedures and the Writer's Guide for Instrumentation and Control Procedures.

Summary of Safety Evaluation: The procedures are unit specific. Prerequisite conditions, TS frequency requirements and acceptance criteria sections were added. Attachments were included to list affected Control Room indications during calibrations. This activity is performed during cold shutdown conditions and testing methodology is not changed from procedures previously approved and performed. (SER 95-030)

14. 1ICP-05.058A-1; 1ICP-05.058B-1; 2ICP-05.058A-1; 2ICP-05.058B-1, Safeguards Timing Relays Calibration Train A and B, Revision 0. (New Procedures)

The tests calibrate the Agastat ETR relays in safeguard racks 1C-156, 1C-166, 2C-156, and 2C-166. The relays are used to sequence loads on to the vital buses after a safety injection (SI) signal or diesel output breaker closure.

Summary of Safety Evaluation: The unit having its relays calibrated must be in cold or refueling shutdown. Only one relay is removed from service at a time during the calibration. The relays cannot cause an accident or malfunction of equipment important to safety since the only purpose of the relays is to start loads after an accident that initiates from a SI signal or a loss of power.

The single component of the shutdown unit whose relay is calibrated is able to be manually started from the Control Room. The procedure is performed during cold or refueling shutdown so the SI timing application for safeguards is not necessary. SI signals are normally blocked during cold shutdown. Service water (SW) availability to the shutdown unit is the limiting case during relay calibration. When the time delay is removed for one of the service water pumps, it is not automatically started when a start signal is received from the shutdown unit. This pump is available to be manually started from the Control Room. The other five service water pumps receive a start signal from the shutdown unit. Each procedure includes a service water operability test for the service water pump relays calibrated by that procedure. This test ensures that the service water pumps start on a start signal from the shutdown unit. All six service water pumps will receive their normal start signals from the operating unit. Therefore, SW system operability for the operating unit is not compromised. The other shared safeguards load (auxiliary feedwater pumps) also start on a SI signal from the operating unit. Thus, the consequences of an accident or malfunction of equipment is not increased.

Administrative controls require that two service water pumps powered from the unit not under test be operating and operable during the time the shutdown unit's service water pump time delay relays are calibrated. This ensures that in the event of a loss of offsite power (LOOP) to the shutdown unit, even if any service water pump is out of service for a reason unrelated to time delay relay (TDR) calibration, and any second pump's automatic start from the shutdown unit is inoperable because its TDR is being calibrated, at least two service water pumps are running, and a third is available for manual start. Two service water pumps are sufficient to supply the normal loads of the operating unit, the loads of the shutdown unit, and an operating emergency diesel generator (EDG). This assumes the failure of an EDG

to start. If an EDG does not fail to start, two service water pumps are running and three others are available. Both of these scenarios assume partial LOOP (to one train only). For complete LOOP to the shutdown unit with a failure of an EDG to start, two service water pumps are running throughout, and one additional pump receives a start signal. Service water pump automatic start from the operating unit on SI or EDG breaker closure is not affected. Thus, service water operability to the shutdown and operating units is maintained. (SER 95-060)

15. ICP-13.009; ICP-13.009-1; ICP-13.009-2, Condensate Storage Tank Level Instruments Yearly Calibration, Revision 0. (New Procedures)

The new procedure was developed from ICP 6.50, "Spec 200 Internals" and ICP 6.59, Calibration Procedure Condensate Storage Tank Level and Service Water Header Pressure Transmitters.

Summary of Safety Evaluation: Since condensate storage tank level is included as TS surveillance requirements, the procedures common to both units were developed. Steps were included to provide a list of prerequisite component and system conditions. Steps were included listing conditions necessary to verify the equipment is returned to service. Acceptance criteria statements were provided to describe conditions and results necessary to satisfy TS requirements. The instruments calibrated are designed to be treated and required to be tested per TS Table 15.4.1-1 and were written to maintain the minimum operable channels requirements of TS 15.3.5, Table 15.3.5-5. The previously used testing methodology was not changed. (SER 95-072)

16. IPSCO FI-91-0211-01, Hot Tapping Into a Storage Cask Possibly Containing Radioactive Gas. (New Procedure)

The procedure facilitates removal of fuel in accordance with RP-8 Part 3. It provides a means to vent pressure below the multi-assembly basket (MSB) valve cover plates prior to removal of the cover plates. This connection is used for collecting gas samples and may be used for complete venting of the MSB if a leak occurs through the valve within the MSB or shield lid seal weld.

Summary of Safety Evaluation: The hot tap fitting and valve are not intended to be an extension of the MSB pressure boundary as licensed in the Safety Analysis Report (SAR). The hot tap fitting allows removal of the structural lid to remove fuel and shall never be in place during normal storage of the fuel in the ventilated storage cask (VSC). The hot tap fitting is designed and tested to withstand the maximum pressure and temperatures in the MSB as called out in the Safety Analysis Report. The maximum pressure assuming failed fuel is 34.6 psig at 850°F. The weld for installing the fitting to the valve covers shall be in accordance with weld procedure WP-17. This is the same procedure used for welding the valve covers in place. The tapping machine and tapping fitting are both pressure tested to 75 psig prior to installation. The system is then pressure tested again to 75 psig after welding the fitting in place prior to drilling the hole through the lid. The valve, tap fitting, drilling machine, packing and other pressure retaining parts are required to be rated to withstand 40 psig at 850°F. This is considered adequate to maintain the pressure boundary for the expected period of time the hot tap shall be in place. The cover plates are removed later in the fuel unloading process and shall not be re-used.

The VSC-24 safety analysis report Section 11.2.1 discusses the hypothetical accident called "Failure of All Fuel Pins with Subsequent Ground Level Break of MSB." The consequences of the MSB failure in this accident is shown to be well below the 5,000 mrem site boundary limit listed in 10 CFR 72.106. The accident includes the failure of the MSB, therefore, the consequences of this accident are not increased by this change. The hot tap fitting is designed and tested for maximum expected pressure.

The Certificate of Compliance (Section 1.1.2) requires a procedure for cask unloading assuming damaged fuel. It states that the Swagelok valves can be used to determine MSB atmosphere prior to removal of the structural and shield lids. For safety reasons, it was determined that the atmosphere and pressure below the valve cover plates should be relieved and sampled prior to removing the valve cover plates in case of valve or shield lid seal weld failure. The hot tap allows for venting and sampling of the atmosphere below the valve cover plates and provides a safe condition for removing the valve cover plates. (SER 95-106)

17. IT-520A, Unit 1 and Unit 2, Leakage Reduction and Preventive Maintenance Program Test of the Safety Injection System, Revision 10. (Permanent)

IT-525A, Unit 2, Leakage Reduction and Preventive Maintenance Program Test of the Safety Injection System (Refueling), Revision 11. (Permanent)

IT-530A, Unit 1, Leakage Reduction and Preventive Maintenance Program Test of the Train A HHSI and RHR Systems (Refueling), Revision 1 (Permanent)

IT-535, Unit 2 Leakage Reduction and Preventive Maintenance Program Test of the RHR System, Revision 10. (Permanent)

IT-535A, Unit 2, Leakage Reduction and Preventive Maintenance Program Test of the Train A HHSI and RHR Systems (Refueling), Revision 1. (Permanent)

The revision eliminates IT-530, "Leakage Reduction and Preventive Maintenance (LRPM) Program Test of the Residual Heat Removal System, Unit 1," and incorporates the LRPM leakage testing of the RHR system into the LRPM test for the safety injection system, IT-520A. IT-520A was renumbered to be IT-530A Train A and IT-530B, Train B. In addition, RHR to containment spray cross-connect valves, SI-MOV-871A&B are tested as boundary valves for the RHR system since the containment spray system is no longer tested in the LRPM program. The same changes were made for the Unit 2 tests, IT-535, and IT-525A.

Summary of Safety Evaluation: An NRC letter dated April 9, 1980, accepts Wisconsin Electric actions to implement NUREG-0578, Item 2.1.6a. This letter states that "A leakage reduction program has been developed and implemented for Point Beach. All systems designed for operation in an accident are included (SI, CS, WGS, sampling and CVCS letdown, charging and holdup tanks.)" The procedure changes do not impact this statement because the test methodology was not changed.

If either train of the RHR system fails the leakage criteria of TS 15.4.4.IV.B, "The maximum allowable leakage... ...shall not exceed two gallons per hour," the plant enters a 72-hour LCO because the train is rendered inoperable. Actions are taken to identify the source of the leakage and correct the failure. This test is performed shortly before the scheduled shutdown in order to ensure that if an LCO would be entered, it would coincide with the scheduled shutdown.

The 2 gpm hydrostatic pump is used to pressurize the RHR system. If pressure cannot be maintained at 350 psig, the test is aborted. If the 2 gph leak rate is exceeded, and it appears to be cross-train leakage between the RHR cross-connect valves (2RH-716A&B, 1&2 RH-716C&D), the opposite RHR train is pressurized to near 350 psig through the suction pressure indicator connection for PI-653A&B using RH-V-02, 02A, and 02B for Train B (RH-V-01, 01A, and 01B valves for Train A). This isolates the cross-train leakage path. The opposite train is operable while pressurized. A dedicated operator is available to ensure the vent valves are shut within 10 minutes if the operable train is needed to respond to a plant incident.

The seismic integrity of the vent valves is not impacted by the connection of the hydrostatic test rig since a flexible hose is used. The test rig is secured by checking the wheels. (SER 95-033)

18. IT-540A Unit 1, "Leakage Reduction and Preventive Maintenance Program Test of Containment Spray System (Annual)," Revision 8. (Permanent)

IT-545A Unit 2, "Leakage Reduction and Preventive Maintenance Program Test of Containment Spray System (Annual)," Revision 8. (Permanent)

The procedures were developed in response to NUREG-0578 (Item 2.1.6a) requirements to monitor and control leakage from systems which could be used with radioactive liquids following an accident. Since the LRPM program was developed, emergency operating procedures have changed regarding containment spray operation. The present EOPs do not align containment spray in the containment sump recirculation mode, although the physical capability exists to do so.

Summary of Safety Evaluation: Since the containment spray system is not expected to be operated with highly radioactive primary system fluid in the event of a design basis accident with core damage, it does not contribute to a radioactive release following an accident. Therefore, there is no need to continue to perform LRPM testing. The SI-871A&B interface between the RHR system and the containment spray system now become part of the RHR boundary per the LRPM testing. These valves are included in LRPM testing for the RHR system. The containment spray system also interfaces with the SI system at the SI mini-recirculation lines. The SI-859A&B check valves and the SI-864A&B manual valves provide the boundary between the SI and containment spray mini-recirculation lines. They are tested for leakage in IT-510B and IT-515B.

NRC letter, dated April 9, 1980, accepts Wisconsin Electric actions to implement NUREG-0578, Item 2.1.6a. The letter states, "A leakage reduction program has been developed and implemented for Point Beach. All systems designed for operation in an accident are included (SI, CS, WGS, sampling and CVCS letdown, charging and holdup tanks)."

The test is not being canceled because it satisfies the ASME Section XI 40-month functional test of Class 2 piping, and the ASME Section XI 10-year hydrostatic test, although it is no longer required for the leakage reduction program. (SER 95-034)

19. IWP 91-116*X1, 1A-04 Bus Rework, Revision 0. (New Procedure)

IWP 91-116*X2, Old 1A-06 Bus Control Cable Disconnect, Revision 0. (New Procedure)

IWP 91-116*X3, Safeguards SI and SW Autostart Signal Installation and Testing/D28 T-MOD, Revision 0. (New Procedure)

IWP 91-116*X4, 1X-14 Station Service Transformer Feeder & Control Cable Rework, Revision 0. (New Procedure)

IWP 91-116*X6, 1P-15B SI Pump Feeder And Control Cable Rework, Revision 0. (New Procedure)

IWP 91-116*X7, Testing of 1P-15B SI Pump Feeder and Control Cable Rework, Revision 0. (New Procedure)

IWP 91-116*X8, 1A-06 New Bus Swapover for Feeder and Control Cable Rework, Revision 0. (New Procedure)

IWP 91-116*X9, Testing of the New 1A-06 Bus, 1A52-84 and 1A52-54 Breaker Rework, Revision 0. (New Procedure)

IWP 91-116*X10, G-03 & G-04 LOOP Test, Revision 0. (New Procedure)

IWP 91-116*X12, Scheduling All Minor IWPs During Outage/LCO - Phase 3B, Revision 0. (New Procedure)

IWP 94-086*X1, Installation of 2 Out of 3 Undervoltage Relay Scheme for 1A-05, Revision 0. (New Procedure)

IWP 94-086*X2, Testing of Extended 1A-05 Bus, Revision 0. (New Procedure)

The procedures control emergency diesel generator (EDG) modification work to be done during U1R22. The work completes Phase 3B of MR 91-116 which adds two new EDGs. This phase ties in G-03 EDG as the Unit 1 Train B standby emergency power source. The existing 4160 V Unit 1 Train B bus 1A-06 becomes an extension of the existing 1A-05 bus. A new 1A-06 bus, which is already connected to G-03, is to be supplied from the existing normal offsite power supply bus 1A-04 for Unit 1 Train B safeguards power. Safety injection (SI) pump 1P-15B and transformer 1X-14 (supply for 480 V safeguards bus 1B-04) is disconnected from the old 1A-06 bus and connected to the new 1A-06 bus.

Summary of Safety Evaluation: The output from G-02 EDG to the existing 1A-06 is disabled. G-02 does not provide standby emergency power for any bus during this phase of the EDG addition. G-03 EDG is the standby emergency power supply for 1A-06. G-04 EDG continues to provide standby emergency power for the 2A-06 bus and becomes a backup for the new 1A-06 (G-03 is not a qualified backup for 2A-06). G-01 EDG remains unaffected as the Train A standby emergency power source for both units (1A-05 and 2A-05).

The previously evaluated accidents or transients applicable to the EDG tie-in activities are a loss of offsite power (LOOP), or a loss of electrical load to Unit 2 or a Unit 2 trip. During most of the activities associated with the G-03 tie-in, Unit 1 is defueled.

One exception to this is the 1P-15B SI pump swapover. The SI pump work is done when the pump is no longer required by TS. The breakers are isolated so there is no impact on offsite power to Unit 1. TS 15.3.2.A requires a boric acid injection path be available when fuel is in the reactor vessel. This requirement is met by having a charging pump, the other SI pump, or another boric acid makeup source available.

A second exception is the dc power swapover for G-02 and G-04. This is a simple short-term switching of dc power supply panels to the two EDGs and does not affect equipment in a way that could initiate an accident or cause a trip of Unit 2 since connections are tested to ensure that no electrical fault potential exists. DC system loading for the interim and final configurations is not a concern.

The electrical switching activities to take the 1A-04 and old 1A-06 buses out of service and tie the 1B-03/1B-04/1B-02 buses together, and when 1A-05 is out of service to tie 1B-03 and 1B-04 together, have no impact on Unit 2 or the availability of offsite power to Unit 2. Offsite power is available to Unit 1 through Train A buses 1A-03, 1A-05, 1B-03 during the first bus tie phase and during the second phase of the bus cross tie, offsite power is available through the Train B 1A-04, 1A-06 and 1B-04 buses. When the buses are isolated and the 1B-03/1B-04/1B-02 or 1B-04/1B-03 buses are tied, electrical loading of the buses is procedurally controlled so no malfunction or overload of the 1B-03 or 1B-04 feeder breakers or G-01 can occur.

Commitments regarding station blackout are met during these activities as G-05 remains operable throughout the tie-in process. Appendix R concerns are met through use of procedural controls contained in Operations Standing Order 4.12.7. This includes a twice per shift fire watch while P-38A auxiliary feedwater pump is out of service.

Calculation 95-038 establishes an electrical loading profile for both bus tie evolutions that does not overload the G-01 EDG or the 1B-04 or 1B-03 feeder breakers, if the 1B-03/1B-04 bus tie breaker fails to trip as designed, during a loss of offsite power (LOOP) or accident on Unit 2. The appropriate loading profile controls are contained within procedures in which the bus tie evolutions are performed. The administrative loading limits and additional loads able to be isolated are also included in the procedure which is kept in the Control Room during this work. This includes an hourly log of the 1X-14 or 1X-13 current readings to ensure the administrative limit is met. The administrative limit includes a margin of safety and is below the limit evaluated in Calculation 95-038.

The margin of safety defined in the Basis for TS 15.3.7 and 15.3.4 is not reduced by these activities. The margin of safety for standby emergency power sources is met by entering TS allowed LCOs for G-01 and G-04. The margin of safety for the auxiliary feedwater system is met by entering the TS allowed LCO for P-38A AFP with P-38B and 2P-29 operable. The margin of safety for having safety-related buses tied together is met by having Unit 1 shut down and defueled with electrical load controlled under the applicable LCO. Amendments 161/165 provide a special note to allow a Train A SW pump to be considered operable, running, and powered from alternate shutdown during the 1A-05 bus extension work. This allows Unit 2 to continue operating without a service water LCO. Under these conditions the required equipment redundancy is maintained and the margin of safety is not reduced. (SER 95-036)

20. IWP 91-116*Y2, G-02 Preoperational Testing, Revision 0. (New Procedure)

Phase 3C of the emergency diesel generator (EDG) project places G-02 EDG in service to 2A-05 as the normal standby emergency power supply. IWP 91-116*Y2 performs the functional logic testing of the new Train A power and control cabling to G-02 and its auxiliaries including: diesel room exhaust fan W-12C; diesel room exhaust fan W-12D; diesel room dampers CV-4176, 4177 and 4178; G-02 EDG engine and generator controls; and G-02 EDG alternate output breaker 1A52-66.

The procedure performs the pre-tie-in performance testing of G-02 that includes a fast start test; a load run/sync test; a 24-hour endurance margin test; a load rejection test; a hot restart test; five start reliability tests; and synchronization test.

Summary of Safety Evaluation: The retraining design assures the new systems meet or exceed existing system design requirements. Evaluations of the new power and control cable routings assure that applicable separation requirements are met.

The retraining of G-02 requires that the power supply and damper control circuits for W-12C be switched from 1B-42 to 2P-32. The power supply and damper control circuits for W-12D are switched from 2B-42 to 2B-32. The G-02 dc control power supply is changed from D-02 to D-03. The 120 Vac instrument power supply to C-35 is switched from 2Y-06 to 1Y-06. Evaluations verify the additional loads placed on D-03, 1Y-06 and 2B-32 for the G-02 auxiliaries are acceptable. The EDG loading calculations were revised to reflect these changes.

The functional testing activities performed for the G-02 alternate output breaker 1A52-66 require 1A52-66 to be placed in the "test" position and closed. This renders G-01 inoperable to the 1A-05 bus because an interlock between 1A52-66 and the G-01 normal output breaker 152-60 prevents 1A52-60 from closing. When this occurs, G-01 EDG is placed in a 7-day LCO per TS 15.3.7.B.1 for 1A-05 and 2A-05. Although G-01 will be in an LCO, it remains aligned to 1A-05. Should G-01 start under these conditions, actions may be taken to trip 1A52-66 and allow 1A52-60 to close, if required.

The energization of the G-02 auxiliary loads on Train A buses was evaluated and the loading was determined to be acceptable. Testing of the G-02 output breaker 1A52-66 and the loading of G-02 to 1A-05 requires entry into an emergency power LCO; however, it does not place either unit in a condition that has not been previously analyzed. (SER 95-070)

21. IWP 91-116*Y3, Tie-in of the G-02 Fuel Oil Transfer System, Revision 0. (New Procedure).

IWP 91-116*Y4, Tie-in of the G-01 Fuel Oil Transfer System, Revision 0. (New Procedure)

IWP 91-116*Y3 completes the State of Wisconsin required pressure test for the G-01 and G-02 EDG underground fuel oil piping, connects power and control cables for the Train A fuel oil transfer system for G-02, and aligns and functionally tests the new Train A fuel oil transfer system for G-02. The system provides fuel to G-02 EDG during the preoperational testing performed in accordance with IWP 91-116*Y2.

IWP 91-116*Y4 ties-in and functionally tests the new Train A fuel oil transfer system for G-01 EDG. It also rewires the controls for existing fuel oil transfer pumps P-70A&B. The pumps no longer provide an automatic fuel transfer to G-01 and G-02. The supply piping from P-70A&B to G-01 and G-02 remain in place. This piping is isolated from the new transfer system for each engine with two isolation valves. G-01 is out of service while the work is performed.

Summary of Safety Evaluation: The design and installation of the Train A fuel oil transfer system meets or exceeds the existing system design requirements. The new fuel oil system utilizes the new fuel oil storage tanks and transfer pumps located in the new EDG building to supply fuel oil to G-01 and G-02. The system provides a greater fuel inventory and a Seismic Class 1 storage system. Piping stress and hydraulic calculations evaluated the system configuration.

The new fuel oil transfer system configuration allows a siphon to be established between the storage tank and the G-01 or G-02 day tank if MOV-3931 should fail open. If the storage tank is at its maximum capacity, approximately 800 gallons of oil would spill from the tank vents. The existing vents on day tanks are extended to prevent a fuel oil spill in the event of the motor-operated valve failure. Spilling 800 gallons of fuel oil would not be a significant loss of inventory and does not create a nuclear safety concern. The vents are extended to prevent a release of fuel oil to the environment.

The cables required for Train A fuel oil transfer system were installed per IWP 91-116*T. Evaluations of the new power and control cable routings assure that applicable separation requirements are met. Evaluations also verify the additional loads placed on D-31, 2B-30 and 2B-32 for the new fuel oil system are acceptable. The tie-in of the Train A fuel oil transfer system to G-02 is performed while G-02 is out of service. The tie-in of the G-02 fuel oil transfer system temporarily disables the G-01 transfer system controls. G-01 is out of service during a portion of this work and an emergency power LCO is entered for 1A-05 and 2A-05 per TS 15.3.7. G-03 and/or G-04 is aligned to 1A-06 and 2A-06 during this work to provide standby emergency power to Train B safeguards equipment. The tie-in of the new Train A fuel oil transfer system to G-01 is controlled by IWP 91-116*Y4. (SERs 95-066, 95-067)

22. IWP 91-116*Y5, Safeguards SI/SW and UV G-03 Autostart Signal Installation and Testing, Revision 0. (New Procedure)

IWP 91-116*Y6, G-02 SI Autostart Signal Installation and Testing, Revision 0. (New Procedure)

IWP 91-116*Y7, 2P-15B SI Pump Control Cable Rework and Testing, Revision 0. (New Procedure)

IWP 91-116*Y8, Disconnect and Remove 50D Relay Wiring From 2A52-87 and Test, Revision 0. (New Procedure)

IWP 91-116*Y9, Disconnect and Remove 50D Relay Wiring From 2A52-93 & 2A52-96/Modify White Light, Revision 0. (New Procedure)

IWP 91-116*Y10, G-03 Mini-LOOP to 2A-06 and G-01, G-02, G-03, & G-04 Redundant Unit Start Test, Revision 0. (New Procedure)

IWP 91-116*Y11, Addition of Test Switches to Degraded Grid Follower Relay Circuit and PT Fuses, Revision 0. (New Procedure)

IWP 91-116*Y12, Scheduling All Minor IWPs During Outage/LCO - Phase 3C, Revision 0. (New Procedure)

IWP 91-116*Y13, MCC 2B-40 Power Indication and G-04 Not-in-Auto Circuit Modification, Revision 0. (New Procedure)

IWP 94-087-1, Old Bus 2A-06 Loss of Voltage Relay Modification - Preoutage Plan Unit 2, Revision 0. (New Procedure)

IWP 94-087-2, Bus 2A-05 Loss of Voltage Relay Modification - Defueled Plan Unit 2, Revision 0. (New Procedure)

IWP 94-087-3, Testing of the Extended 2A-05 Bus, Revision 0. (New Procedure)

The procedures accomplish Phase 3C of MR 91-116, for the addition of two EDGs. This phase ties-in G-02 EDG as the Unit 2 Train A standby emergency power source. The old 4160 V Unit 2 Train B bus "old 2A-06" becomes an extension of the existing 2A-05 bus. The "old 2A-06" bus was abandoned in place during Phase 3A tie-in activities. In addition, G-03 EDG is made operable to the 2A-06 bus as an alternate standby emergency power source.

Summary of Safety Evaluation: The previously evaluated accidents or transients applicable to the EDG tie-in activities are a loss of offsite power (LOOP), loss of electrical load to Unit 1 and a Unit 1 trip. Unit 2 is defueled during most of the activities associated with the G-02 tie-in.

One exception to the above is the 2P-15B safety injection (SI) pump control cable rerouting. The SI pump work is done when the SI pump is no longer required by TS and its breakers are isolated. This has no impact on offsite power to Unit 2. TS 15.3.2.A requires a boric acid injection path be available when fuel is in the reactor vessel. This requirement is met by having a charging pump, the other SI pump, or another boric acid makeup source available.

A second exception is the G-03 safety injection/service water/undervoltage start connections and 2B-40 circuit modifications which are done prior to core offload. During this time G-03 and G-04 EDG LCOs are entered. Train A equipment and shared safeguards equipment remain operable. Unit 2 Train B safeguards racks are deenergized and Train A remains energized. This work is done with Unit 2 in cold shutdown. The testing of the G-03 and G-04 EDGs to the 2A-06 bus have no impact on the availability of offsite power.

Some of the control system work has the potential to affect Unit 1 because a portion is done in the Train B safeguards rack 1C-167. This involves connecting an auto start signal to G-03 on a Unit 2 SI signal. Appropriate care is taken when working in these racks to prevent affecting Unit 1. Precautions and double verification steps to ensure that the correct termination point is selected and the jumper for testing the circuit is properly placed. Additional work is done in control boards C-01 and C-02. This work is similar to that previously performed for the EDG tie-ins and appropriate procedural steps are in place to ensure the work can be done without affecting Unit 1. Appropriate precautions minimize the risk of inadvertently affecting equipment important to safety.

A third exception is the preparatory work done for the 2A-05 bus extension. This is done with Unit 2 in cold shutdown and prior to the Train B work. Requirements for the 2A-05 bus extension must be complete or halted prior to the Train B work taking place.

The electrical switching activities first take the 2A-05 and then 2A-06 buses out of service and tie the 2B-03 and 2B-04 buses together, have no impact on Unit 1 or the availability of the offsite power to Unit 1. Offsite power is available to Unit 2 through the Train B buses 2A-04, 2A-06, 2B-04 and 2B-03 during the first bus tie phase and during the second phase of the bus cross-tie, offsite power is available through the Train A 2A-03, "new" 2A-05, 2B-03 and 2B-04 buses. When the buses to be worked are isolated and the 2B-04/2B-03 or 2B-03/2B-04 buses are tied, electrical loading of the buses are procedurally controlled so no malfunction or overload of the 1B-03 or 1B-04 feeder breakers, G-04, or G-02 can occur.

Commitments regarding station blackout are met during these activities as the G-05 gas turbine, remains operable throughout the tie-in process. Appendix R concerns are met through use of procedural controls requiring fire rounds when Appendix R equipment is out of service. (SER 95-103)

23. NP 3.2.2, Primary Water Chemistry Monitoring Program. (Temporary)

NP 3.2.2 was temporarily changed to increase the upper administrative limit for safety injection accumulator boron concentration from 2500 ppm to 2700 ppm. This reflects that accumulator boron limits are applicable when the accumulators are required to be operable.

Summary of Safety Evaluation: The accumulators are a passive safety system for accident mitigation. They operate when RCS pressure decreases below the accumulator pressure, allowing flow from the accumulator to the cold leg via a check valve. They provide a source of borated water for core cooling and reactivity control. The structural integrity and corrosion are not adversely affected by the slightly higher boron concentration.

The increased boron concentration does not adversely affect the capability of the accumulators to perform their safety function. The accumulator TS pressure, volume, and minimum boron concentration is not changed. There is no TS limit on maximum boron concentration in the accumulators. The increase in boron concentration is a benefit for reactivity control. The accumulator internal pH differs from 2700 ppm versus 2500 ppm (4.50 versus 4.46). This slight change does not adversely affect corrosion. The solubility of boric acid in water at 0°C is greater than 4500 ppm, so boron precipitation is not a concern.

The slight increase in boron concentration has a negligible effect on the fluid flow properties, so accumulator flows are not impacted. The amount of nitrogen that may be dissolved in water is primarily dependent on temperature and pressure. The change in boron concentration has a negligible effect on the amount of dissolved nitrogen. The accumulator level indicators are capacitance probes. The change in conductivity due to level changes is much greater and overrides small changes in the fluid conductivity. The level instruments continue to function accurately.

The accumulator's discharge is also a factor in the containment sump pH during recirculation. Calculations show that accumulators having 2700 ppm boron (rather than 2500 ppm boron) have such a small impact on boron concentration in the sump (approximately 11 ppm increase out of >2400 ppm), that sump pH does not change measurably and is within an acceptable range for recirculation. Thus, long-term core cooling post-LOCA is maintained. Because the sump pH is maintained within an acceptable range, sump water can be used to supply containment spray. The spray system is not adversely affected and environmental qualification requirements are met.

An NRC safety evaluation report dated December 24, 1975, specifies that core deluge flow needs to be established within 14-hours of a LOCA to eliminate concerns of post-LOCA boron precipitation. This time limit continues to be met as directed by EOP 1.2 for a small break LOCA, where the accumulators may or may not partially discharge. For a large break LOCA, core deluge flow occurs within 14 hours by plant design without operator action. The slight increase in accumulator boron concentration does not change the conclusions of this SER. Accumulator discharge is small compared to the RCS, CVCS, and SI injected volumes. (SER 95-116)

24. OI-104, Spray Additive Tank Level Adjustment and Recirculation, Revision 2. (Permanent)

The revision provides a method to drain a volume of sodium hydroxide (NaOH) from a spray additive tank to the neutralizing tank and to discharge the sodium hydroxide after it is neutralized. It provides instructions on equalizing levels between the two tanks.

Summary of Safety Evaluation: The 30% NaOH solution is drained from the sodium hydroxide tanks to the neutralizing tank. The tanks are drained from tank drain valves to the neutralizing tank utilizing temporary hoses, fittings and a pump currently used for tank recirculation operations. Since the tanks are drained from existing drain valves to a tank which is open to atmosphere and the drain hose could not create a siphon, the draining operation does not require a temporary modification. The temporary hoses, fittings and pump were previously evaluated for acceptability in handling the NaOH fluid. After neutralizing the tank contents, Chemistry samples the tank and prepares discharge permits to meet TS and WPDES requirements.

A jumper hose is installed so the release can be continuously monitored by retention pond monitor RE-230. The valve lineup isolates retention pond flow from RE-230 and monitors flow from the neutralizing tank discharge line. The RE-230 high alarm setpoint is conservative for this application since the setpoint is based on a retention pond overboard flow of 2670 gpm. The neutralizing tank discharge only flows at 200 gpm.

Procedural requirements meet the intent of the FSAR regarding liquid waste disposal system releases, even though this release is not from the liquid waste disposal system. Valve WT-52 is locked shut while the NaOH solution is drained to the tank and neutralized until discharge permits are completed. During the discharge operation, RE-230 is placed on a continuous trend in the Control Room and a dedicated operator with radio contact is stationed at the neutralizing tank discharge valve. If RE-230 goes into high alarm, the dedicated operator secures the release by shutting neutralizing tank pump discharge and overboard valves WT-50A&B and WT-51 & WT-52.

The temporary drain hoses are routed away from equipment as much as possible. Barricades are established around the hoses to enhance personnel safety. Double hose clamps are used on the hose connections. Operators are stationed in the primary auxiliary building and the water treatment area while the NaOH tanks are drained. They watch for leaks so the spray additive tank valve can be shut if necessary. A caustic neutralizing agent is available. Fire rounds are established as required by TS and Appendix R for the fire barriers degraded by the temporary hose.

Prior to draining the NaOH to the neutralizing tank, the neutralizing tank is filled to a level of 50% water to dilute the NaOH upon addition. The fluid is neutralized after the NaOH is added. The caustic addition and neutralization in the neutralizing tank are routine evolutions, and fumes generated are not a concern. Calculations show that the tank contents remain below 100°F during the evolutions. This is acceptable since the tank has a short-term design temperature of 120°F and the radiation monitor has a 122°F limit. (SER 95-003)

25. OI-109, TM 95-029, SFP Cask Laydown Area Vacuum Cleaning, Revision 0. (New Procedure)

Procedure OI-109 and temporary modification TM 95-029 are used to vacuum clean accumulated silt and small debris from the spent fuel pool (SFP) cask laydown area using a submersed pump tied into the SFP demineralizer and filter system. The procedure temporarily suspends purification and skimmer pump operation in the SFP as the cask area needs to be cleaned in preparation for dry cask storage moves. TM 95-029 attaches a hose from the submersed pump discharge into the SFP demineralizer system by valve SF-817C. The threaded cap downstream of SF-817 is removed and a hose connection is attached.

Summary of Safety Evaluation: The valve lineup maintains SFP cooling, but suspends normal purification flow. Flow is instead delivered to the filter and demineralizer from the submersed vacuum pump via valves SF-817C, SF-817B, and SF-795. Control of the flow is the same as that used to control demineralizer flow via SF-812A. Flow is limited to 60 gpm; the design flow of the demineralizer. The pump has a 480 V (non-safeguards), 5 hp motor with a shutoff head of 130 psi. The setpoint for the two relief valves in this system is 132 psi. An operator is stationed at the discharge pressure gauge of P-33 to monitor system pressure during the initial lineup. The valve lineup is performed in an order so the system relief valves do not see the pump deadhead pressure. The system lineup does not contain any automatic valves that may inadvertently shut. The discharge hose is rated for a working pressure of 150 psi, a burst pressure of 400 psi, and has been tested to 225 psi. The discharge hose is double clamped to the connector. Should system leakage occur, the pump may be shut off and the system isolated or the suction hose removed from the pool.

The installation of the pump assembly into the SFP utilizes the single-failure proof PAB crane. Its rigging meets appropriate standards and safety margins. The 400 pound assembly is not a heavy load. Loads of this nature are routinely moved in and over the SFP (a fuel assembly weighs about 950 pounds).

FSAR 14.2.1 states the SFP rack structure protects stored fuel from laterally bending loads. The maximum expected load to be moved in the SFP with a potential for dropping onto the storage racks is approximately 1000 pounds. This value used in the design of fuel handling equipment establishes limits for inadvertent axial loads. The pump assembly weighs about 400 pounds. Therefore the consequences of a drop of the pump assembly are bounded by those analyzed in the FSAR and subsequent safety evaluation reports.

The consequences of a spill of SFP water are not increased because the system may be isolated and the SFP water activity is not increased. (SER 95-065)

26. OM 3.26, Use of Dedicated Operators, Containment Integrity Issues, Revision 0. (New Procedure)

OM 3.26 replaces standing order PBNP 4.12.54. The administrative procedure clarifies the requirements for using dedicated operators for plant evolutions.

Summary of Safety Evaluation: A thorough search of the FSAR, TS, and plant technical files regarding containment isolation was performed to determine the safety analysis time limit associated with the automatic containment isolation system. No specific requirements were found. FSAR Table 7.5-1, "General Operating Time Requirements for Environmental Qualification of Electrical Equipment," lists "times to operate" for various plant equipment. The table defines time to operate as the time after an accident in which it is expected that the item will have completed its safety function. The table lists 10 seconds for air-operated containment isolation valves (CIVs). While this can be misinterpreted to mean that the air-operated containment isolation valve must reposition within 10 seconds after an accident, this is not the case.

NUREG-0800 Standard Review Plan 6.2.4 on "Containment Isolation Systems" and ANS N271-1976, "Containment Isolation Provisions for Fluid Systems," list one minute as the maximum time that should be allowed for a power-operated containment isolation valve to reposition in response to a signal to shut. The standards provide a reasonable time limit that can be used to provide guidance to dedicated operators stationed in containment isolation applications. Additionally, the preceding standards also discuss that longer operating times for CIVs may be justifiable on an individual basis where evaluation of the specific situation demonstrates that the longer operating time does not result in a significant increase in offsite dose under accident conditions. TS 15.3.6.A.1.b and its Basis allow the use of dedicated operators for containment isolation flow paths provided the operator is in communication with the Control Room and is stationed at the valve controls. This allows the penetration to be rapidly shut when a need for containment isolation is indicated. While the one minute guidance is desirable, the intent is to have the operator shut the penetration as quickly as possible. (SER 95-010)

27. OP-3A, Normal Power Operation to Low Power Operation, Revision 34. (Temporary)

The temporary change allows auxiliary feedwater system testing to 20-30% power irrespective of the initial boron concentration. The procedure currently allows testing at 20-30% power when the RCS boron concentration is (≤ 200 ppm). Auxiliary feedwater system testing was done when reactor power was < 2%.

Summary of Safety Evaluation: Reactivity changes because of this testing are less severe and easier to control at a higher power level. The boron concentration is not a limiting concern. Thermal fatigue of the feedwater nozzles to the steam generator is also reduced at higher power. The thermal fatigue issue was addressed in SER 91-022.

The magnitude of the reactivity change depends on the moderator temperature coefficient (MTC) and the amount of hot main feedwater being added to the steam generator. At a higher power level there is more hot main feedwater mixing with the auxiliary feedwater which reduces the RCS temperature reduction and minimizes the reactivity changes. MTC typically varies from +5 pcm/°F at beginning-of-life, 1600 ppm boron, and zero power to -30 pcm/°F at end-of-life, 0 ppm boron and full power. Boron concentrations >200 ppm do not produce the most limiting reactivity changes. The largest reactivity additions occur at end-of-life (MTC = -30 pcm/°F) and at low power.

Reactivity is more difficult to control at low power, (< 2%,) due to the xenon buildup occurring as power is reduced during the procedure. Xenon buildup is of greater concern at low boron concentrations because the boron concentration cannot be further reduced to overcome xenon poisoning.

The change does not increase the number or frequency of auxiliary feedwater system tests. The test could not increase the reactivity of the core as in the "Reduction of Feedwater Enthalpy Incident" evaluated in FSAR 14.1.6. The changes do not increase the consequences of the potential reactivity addition. Performing the test at a higher power level reduces the magnitude of the reactivity addition and therefore reduces the likelihood of a departure from nucleate boiling (DNB). Greater control of the reactivity allows the operator to reduce the consequences of a reactivity addition during the test. Maintenance and reliability of the equipment in use at the two power levels is similar and does not increase the probability or consequences of a malfunction. The probability of a fatigue failure of a steam generator feedwater nozzle is reduced because of a change in the feedwater temperature, resulting from auxiliary feedwater flow, being less at higher power levels.

The higher power makes the reactivity changes from auxiliary feedwater system testing easier to control regardless of the initial boron concentration. Thermal fatigue from auxiliary feedwater testing is reduced at higher power. Therefore, it is beneficial to perform the testing at 20-30% power for any initial boron concentration. (SER 95-035)

28. ORT-3A, Safety Injection Actuation with Loss of Engineered Safeguards AC, Unit 2. (Permanent)

The change addressed the expanded scope resulting from the G-02 EDG tie-in and the qualification of G-03 to Unit 2. The test is used as part of the G-02 EDG operability determination and G-01 operability to the new extended 2A-05 bus. It also shows operability of G-03 to the 2A-06 bus and is the normal test of G-04 to the 2A-06 bus.

Summary of Safety Evaluation: ORT-3 demonstrates ECCS and EDG operability as described in the FSAR and as required by TS. This test is performed during U2R21 with the unit in cold or refueling shutdown and appropriate TS LCOs are entered for EDGs as well as shared safeguards equipment.

Because of restrictions associated with the test (no other LCOs entered, no safeguards systems work or testing, no fuel motion) and because Unit 2 is in cold or refueling shutdown, there are no previously evaluated accidents which could be directly affected by the changes, nor is the availability of offsite power affected. While Unit 2 safeguards buses are de-energized and restored, this has no effect on the operating unit. The test does not initiate any accidents previously evaluated for the at-power unit. G-01 and G-02 are normally Unit 1 EDGs and during this test on Unit 2, standby power LCOs are sequentially entered for Unit 1. G-04 is normally a Unit 2 EDG and no LCO for Unit 1 is entered. The shared safeguards loads

powered from 2B-04 remain available to Unit 1 except for the short period that 2B-04 undervoltage load stripping is verified. Equipment LCOs are entered for the shared safeguards equipment. G-02 is the normal EDG for Unit 2 and no LCO is required as ORT-3 is the qualification test for G-02.

The test is set up so RHR flow is not interrupted if fuel is in the reactor vessel. The testing is performed one train at a time. The Unit 1 spent fuel pool (SFP) cooling pump is in operation. The major change is that G-01, G-02, G-03, and G-04 are loaded. This does not affect or increase the probability of a loss of offsite power.

The incorporation of degraded voltage and undervoltage testing with the load shedding and restoration test does not test the EDG in a manner different from its design function. G-02 is not required to respond to accidents while the unit is shutdown, but is available as a power source to the service water pump and G-01 fan that are the shared safeguards loads from Unit 2 Train A. During the G-02 and G-03 tests, a single largest load rejection test is performed. During the test, the safety injection (SI) pump is stopped and after a 15 second coastdown, it is restored. This tests the EDGs capability to handle the load swing. The 15 second coastdown is a safety margin which ensures no damage to the SI pump motor occurs. In addition, the CS pump is manually started to provide an additional load. This is started 10 seconds after EDG breaker closure. This is consistent with FSAR EDG loading criteria and does not affect EDG or CS pump operability.

The changes do not create new accident initiators. Unit 2 will be in cold or refueling shutdown. The Unit 2 SFP cooling pump is in operation to prevent a loss of SFP cooling. The test is performed by train and setup so RHR cooling is not interrupted if fuel is in the reactor vessel. Offsite power is available during the test and the tested components are available with manual or automatic action. This test is required by TS, and as such, is part of the normal equipment use. Equipment is used within its design basis. (SER 95-113)

Summary of Safety Evaluation: Because of the restrictions associated with the test (no other LCOs entered, no safeguards systems work or testing, no fuel motion), and because Unit 2 is in cold or refueling shutdown, there are no previously evaluated accidents which could be directly affected by the changes. One exception to these restrictions is procedure ICP 2.1, "Reactor Protection and Safeguards Analog Channels I through IV." Performance of this procedure on the opposite operating unit has been evaluated to have no significant impact on the operation of Unit 1 and does not increase the probability of a trip or safeguards actuation for Unit 1. Availability of offsite power is not affected by these changes. (SER 95-113-01)

29. ORT-3B, Safety Injection Actuation With Loss of Engineered Safeguards AC Unit 1, Revision 28. (Permanent)

The procedure scope was expanded as a result of the G-03 EDG tie-in and the qualification of G-04 to Unit 1. The revised test was used as part of the G-03 operability determination and G-04 operability to Unit 1. The changes for G-03 and G-04 testing for Unit 1 included:

- Having G-03 pick up actual loads, running the B safety injection (SI) pump with 1000 gpm recirculation flow, running B containment spray (CS) pump on mini-recirculation, and running 1W-1C1 and 1W-1D1 containment accident fans;

- Rejecting the largest single load component (P-15B) while the EDG is loaded and its subsequent restoration (this single load rejection test meets the Regulatory Guide 1.9, Revision 3 and ANSI/IEEE STD 387-1984 Section 6.3 criteria commitments);

- Performing the degraded voltage and undervoltage testing in conjunction with the load shedding and restoration test (this reduces the test duration). The undervoltage and degraded voltage testing was performed twice during the G-03 portion of the test in order to test the components. The revisions are identical to the G-04 EDG tests performed during U2R20 with the exception of an additional auxiliary feedwater pump (AFP) and service water (SW) pump (normal 2B-04 loads) which were loaded onto G-04;

- Testing G-04 EDG to the 1A-06 bus on an undervoltage coincident with an SI signal. G-04 sequences on actual loads but the load rejection test is not required because G-04 was already tested. This test qualified G-04 and the 1A52-86 breaker to the 1A-06 bus. G-04 will be in a TS LCO to Unit 2 due to inability to supply Unit 1 and Unit 2 accident loads simultaneously. Also, the load shedding and restoration tests, and the undervoltage and degraded voltage relay tests, are not required during the G-04 portion of ORT-3 because it will have been completed as a portion of the G-03 testing;

- After the G-04 test to the 1A-06 bus, G-03 was again loaded onto the 1A-06 bus on an undervoltage and sequence on loads. This qualifies the interlock between the 1A52-80 and 1A52-86 EDG output breakers. This interlock prevents loading of an EDG onto a bus when another EDG output breaker to that bus is closed.

Summary of Safety Evaluation: The test is conducted during cold shutdown in accordance with TS requirements and applicable LCOs. Because the unit is in cold shutdown, there are no accidents which could be directly affected. The availability of offsite power is not affected. These tests do not initiate any accidents previously evaluated for the at-power unit. There is out of service time for some equipment; however, applicable LCOs are entered. G-04 is normally a Unit 2 EDG, so during this test in order to qualify it to Unit 1, an LCO is entered. G-03 is specific to Unit 1 and is not tested to Unit 2. The only shared safeguards load is one SW pump. This shared load remains available to Unit 2 except for the short period that 1B04 undervoltage load stripping is verified.

The test is set up so RHR flow is not interrupted. The testing is performed one train at a time. Components are tested commensurate with their design function. Therefore, the changes to this test do not increase the probability of occurrence of a malfunction of equipment important to safety.

The changes do not create any new accident initiators. The changes do not increase the probability of occurrence of an equipment malfunction of a different type. Unit 2 is in cold shutdown. During the test, both trains for RHR are operable. The test is performed by train and set up so RHR cooling is not interrupted. Offsite power is available during the test and tested components are available with manual or automatic action. Subsequent to the G-04 test, offsite power is restored to the 1A-06 bus and G-03 is again loaded onto the bus. By restoring G-03 to supplying 1A-06, its closure circuit is verified to be operable since previous G-04 output breaker closure because of 1A-06 defeats G-03 breaker closure because of a design interlock. Procedural controls and the design interlock prevent the possibility of both EDGs from closing in on the bus at the same time. (SER 95-056)

30. PBTP-039, Operation with No. 5 Feedwater Heater Bypass Valve Open, Revision 0. (New Procedure)

PBTP-039 describes operating Unit 2 with the high pressure feedwater heater bypass valve, 2CS-155, open. Operating Unit 2 in this configuration allows approximately one-third of the feedwater to bypass the high pressure feedwater heaters (FWHs). The resultant feedwater inlet temperature to the SG is about 27°F lower (approximately 402°F). Although this introduces inefficiency, it allows reactor power to be returned to 100%. This provides an additional output of 4-5 Mwe.

Summary of Safety Evaluation: Certain limitations are in effect during opening of the valve and the power ascension to follow. The bypass valve is slowly opened to allow plant parameters to change in a slow and controlled manner. Existing primary differential limits are maintained. Once full reactor power is reached, the resultant differential temperatures are analyzed to determine if additional actions are required. At power levels above 98.5%, the load increase is limited to ≤3% per hour as specified for unconditioned fuel.

Operating with the No. 5 FWH bypass valves open results in a lower final feedwater temperature; however, this does not increase the probability of an accident analyzed in the FSAR. The lower feedwater temperature is a concern from a thermal stress perspective but is bounded by the analysis for initiation of cold (70°F) auxiliary feedwater flow. The components of concern are the feedwater inlet nozzle to the SG and feed ring itself. The resultant feedwater temperature (approximately 402°F) is well above the temperature of the auxiliary feedwater. This is a steady-state condition rather than the thermal transient associated with auxiliary feedwater initiation and testing.

The inlet to CS-155 is radiographed to ensure no metallic debris in the piping could bind the feedwater regulating valves. This precludes malfunction of this type, so the probability of a malfunction is not increased.

The margin of safety is not changed. The existing primary differential temperature limits and reactor power limits are maintained. Reactor power is reduced prior to the evolution to allow adequate margin to the 100% power limit. The bypass valve is slowly opened to ensure an overpower condition does not occur.

During the valve opening and power ascension phases several parameters are monitored (reactor power, differential temperature limits, main steam pressure, first stage pressure and feedwater temperature). Additionally, several precautions are adhered to: Existing OPΔT and OTΔT; 100% reactor power; Tref and RTO limits. Once the reactor is returned to 100% power, the potential for nuclear instrumentation shadowing is evaluated (e.g., temperature error effects as a result of changes to first-stage pressure and changes to MSR steam throttle settings). The potential for flow accelerated corrosion on the bypass line is also evaluated. The final total main steam flow is evaluated and compared to normal 100% steam flow. This ensures the protection capabilities of the crossover steam dump system are not exceeded. (SER 95-100)

Summary of Safety Evaluation: The steam generator tube rupture accident analysis postulates a worst case tube shear. This results in a reactor trip from underpressure within seconds, followed by tripping of the turbine throttle valves quickly thereafter. This results in the SG being at no load conditions (1000 psig) within seconds after accident initiation. The major factors in this accident are the 30-minutes which are assumed to be needed to depressurize and the primary coolant activity levels. The SG pressure at the start of the accident is not a significant factor. The lowered SG pressure in this condition does not affect the results of this accident analysis. (SER 95-100-01)

31. RP-7, Dry Cask Loading and Storage, Revision 0. (New Procedure)

RDW19.0, Transportation of the VSC to the PAB Truck Access, Revision 0. (New Procedure)

RDW19.2, Loading and Placing the VSC Into Storage, Revision 0. (New Procedure)

AOP-8G, Multi-Assembly Sealed Basket (MSB) MSB Transfer Cask (MTC) or Ventilated Concrete Storage Cask (VSC) Drop or Tipover, Revision 0. (New Procedure)

These are operating procedures associated with the independent spent fuel storage installation (ISFSI). The operating procedures are utilized during the dry run of the loading process, which is required to be performed before an actual VSC is loaded. Procedure enhancements identified during the dry run are made before the first cask is loaded.

Summary of Safety Evaluation: The selection of fuel assemblies to be stored in the VSC is limited by the requirements contained in the Certificate of Compliance. Since these limits require that the fuel moved during the loading of the MSB is a minimum of 5 years old, and a maximum initial enrichment of 4.2 wt% U-235, and only one assembly can be moved into the MSB at a time, the consequences of a fuel handling accident while moving that fuel assembly are bounded by those analyzed in the FSAR and subsequent safety evaluation reports. The identity of each fuel assembly is independently verified prior to placement into the MSB. This ensures that only selected fuel assemblies are loaded. Therefore, there is no increase in consequences of a fuel handling accident during loading of the MSB.

The crane is seismically designed with and without a load. An additional evaluation for the PAB structure during a seismic event with the crane loaded verified that the PAB structure is acceptable. The MTC, MSB, VCC, and lifting yoke are classified as important to safety and are covered under 10 CFR 50 Appendix B. Before using the PAB crane to lift a MSB/MTC, a complete crane inspection is performed. The inspection ensures the crane continues to meet its original design specifications. The inspection is performed by the original crane manufacturer.

The slow speed of the transporter combined with the redundant braking system ensures that the transporter remains on the direct route. The transporter is also tested with a weight at least equal to the weight of a loaded cask. A clear path for the transporter to and from the PAB is ensured before the VSC is moved by verifying that the roadway is clear, free of obstructions, and that the north gate is operable. These actions prevent delays in transporting the VSC. (SER 94-041-01)

32. RP-7 Part 5, Remove the Multi-Assembly Sealed Basket (MSB) and the MSB Transfer Cask (MTC) from the Spent Fuel Pool, Revision 2. (Permanent)

This revision places the MSB shield lid on the SFP divider wall so long slings can be attached. This prevents the immersion of the PAB crane hook into the SFP. Immersing the hook presents contamination problems for both the hook and the SFP. The sling changeout is done with the water level in the SFP below the level of the divider wall but above the minimum level required for the dry cask procedures.

Summary of Safety Evaluation: The accident or incident considered in this evaluation is the potential damage to the SFP, SFP liner, or SFP contents caused by the shield lid. Since the MSB shield lid weighs approximately 4500 pounds, if it dropped into the SFP from the divider wall, it could damage fuel stored in the racks. The only way the lid could drop into the pool would be if an earthquake occurred when the lid is unsecured from the crane. It is postulated that the earthquake could cause the lid to slide off the divider wall. Administrative controls ensure that the amount of time the lid is unsecured is minimized. The procedure specifies that the slinging changes be performed as a continuous action, without interruption.

Calculations determined that due to the shape, material of construction, and weight of the shield lid, it would not slide off the divider wall during the design basis earthquake (DBE). The calculation also shows that the SFP liner will not be damaged by sliding of the shield lid.

Probabilistic safety assessment (PSA) has determined that the probability of the DBE occurring with the lid unsecured is 3E-7 and the probability of the SSE occurring with the lid unsecured is 1E-7. These probabilities are based on the assumption of loading 8 casks per year.

A new type of accident could be created if the PAB crane moves horizontally before all three slings from the shield lid are attached to the hook. In this case the lid could be dragged from the divider wall and could possibly damage fuel in the racks. To prevent this, a precaution ensures three lifting eyes on the shield lid are attached to their appropriate slings before the slings are attached to the crane hook. (SER 95-074)

33. RP-8, Unloading the Multi-Assembly Sealed Basket (MSB), Parts 1, 2, 4, and 5, Revision 0. (New Procedures)

These are MSB unloading operating procedures (the RP-8 series, except for RP-8, Part 3.) The procedures do not present new tasks or risks different than the loading procedures. Therefore, the evaluations performed for the loading procedures are valid for the unloading procedures, except for RP-8, Part 3. RP-8, Part 3 describes cooling, reflooding, and opening the MSB, and was separately evaluated.

Summary of Safety Evaluation: The equipment used to handle fuel assemblies in the SFP is the existing fuel handling equipment. Existing fuel handling procedures are used as a basis for specific MSB loading and unloading procedures. Fuel handling personnel are trained prior to actual fuel movement. Identical equipment used in moving fuel within the SFP is used to load and unload the MSB in the cask loading area. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is not increased.

A time limit greater than 47 hours is allowable if Kw loading of the MSB is less than 24 Kw. The actual time limit for each MSB is calculated during the fuel selection process. This limit is then tracked in the loading and unloading procedures to ensure that the MSB is either drained or returned to the SFP before the time limit expires. (SER 94-041-02)

Summary of Safety Evaluation: This evaluation also addresses RP-8 Part 3 that is entered when sampling the multi-assembly sealed basket (MSB) it is determined failed fuel is present. If the sample detects the presence of failed fuel, venting and purging activities are performed after the MSB is returned to the VCC. This is done because in some postulated scenarios the quantity of fission product gases present requires a very slow purge rate which could take several days. In that case the MSB is returned to the VCC to ensure it is adequately cooled. After purging is complete, water is introduced into the MSB at a controlled rate until the MSB and its contents are cooled. The MTC/MSB is placed into the SFP, the shield lid is removed, and the fuel assemblies are unloaded.

RP-8 Part 3 does not affect off-normal events or accidents that result in a radiological release. There are no activities in RP-8 Part 3 which affect the ability of safety-related components of the dry storage system to perform its intended functions.

An additional accident of a different type which could be created by the performance of RP-8 Part 3 is the overpressurization of the MSB when the MSB is reflooded with water after it has been vented and purged. The overpressurization could occur if the water was introduced into the MSB faster than the vent line could relieve the steam which is produced by the hot components within the MSB. The consequences of such an accident would be no worse than the analyzed accident, "Rupture of All Fuel Pins with Subsequent Ground Level Breach of MSB." This accident is prevented by pumping the reflood water into the MSB at a rate determined in the reflood analysis. Procedural steps ensure that the flow rates and volumes are in accordance with those required by the analysis to maintain the MSB pressure within acceptable limits. Sufficient equipment is available in the vacuum drying and reflood system to perform these tasks.

Another accident of a different type which could be created by the performance of RP-8 Part 3 is the dilution of the SFP during the initial phase of reflooding. Dilution could occur because of the steaming of the water that is initially introduced into the MSB. It is assumed that the boron is not carried over with the steam, and remains in the MSB. The condensed water from the reflood is returned to the SFP. The amount of pure water which returns to the SFP is conservatively estimated to be 600 gallons. In accordance with the reflood analysis, to ensure that the boron concentration of the water pumped to the MSB during the reflood evolution is maintained above the Certificate of Compliance required limit, the concentration is verified to be adequate, and increased if necessary, before the reflood evolution begins. The amount of the increase is enough to account for the dilution by the 600 gallons. All dilution paths (DI water sources) are double verified and isolated shut during the reflood.

The use of a hot tap to penetrate the structural lid ensures that the confinement boundary has not failed and minimizes potential exposure involved in removal of the access ports.

The VCC shield ring installed during the venting and purging process provides additional shielding from radiation streaming up the MSB/VCC inner liner annulus. This is in accordance with the VSC system design and the shielding analysis.

RP-8 Part 3 does not create new environmental hazards. The methods employed in the procedure actually reduce the consequences to the worse case fuel failure to acceptable levels. Activities performed in the procedure are done in the primary auxiliary building, where the spread of contamination and the release of fission product gases can be controlled. (SER 94-041-03)

Summary of Safety Evaluation: A complete inspection of the primary auxiliary building crane ensures the crane continues to meet its original design specifications. The inspection was performed by the original crane manufacturer. During the inspection safety features of the crane were tested. The crane overspeed device was tested separately from the vendor inspection. (SER 94-041-04)

34. SLP-1, Items Lifted by Containment Polar Crane, Revision 6. (Permanent)

The change incorporates the requirements for use of reactor vessel head shielding during refueling and maintenance activities. The shielding reduces personnel radiation exposure levels.

Summary of Safety Evaluation: The reactor vessel head shielding system was designed so installation and removal is a routine process. The changes are necessary since the lead shielding blankets qualify as a "heavy load" per NUREG-0612 so movement must be carefully controlled.

MR 90-163 implemented the design for the shielding and support structure and specified the shielding storage requirements in containment during plant operation. RMP 96 controls the sequence in which the shielding is installed and removed during refueling and maintenance activities so a lift of the reactor vessel head is not made when the shielding is installed. This control is necessary so the capacity of the polar crane is not exceeded during reactor vessel head lifts.

Movement of the lead shielding is in accordance with NP 8.4.7. This administrative procedure implements the requirements of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants." The defined safe load paths for the lead shielding are similar to other equipment frequently used during refueling and maintenance activities.

The radiation doses received by the general public and by plant workers as a result of a malfunction are not increased. No fission product boundaries, (e.g., fuel cladding, containment and reactor) are degraded by the changes. Movement of lead shielding along an approved safe load path is similar to other previously approved equipment movements and does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

Adding the weight of the lead shielding blankets and approved safe load path routes do not affect any of the equipment, components or systems important to safety as defined in TS. Thus, there is no reduction in the margin of safety as defined in the TS. (SER 95-040)

35. STPT-2.4, Safeguards Sequence Time Delay Relay Setpoints, Revision 2. (Permanent)

The safeguards time delay relays were changed from Agastat Series 2400 pneumatic to electronic relays (ETR). The relays were replaced because of difficulty in setting the pneumatic relays to the desired time, and because of variation in timing between the first and subsequent timings of the relay. The ETRs are not expected to have these shortcomings.

Summary of Safety Evaluation: To take full advantage of the available tolerance range, and the ETR's improved repeatability, it is desirable to set the relays at the middle of the allowable tolerance in the FSAR. This results in a nominal setpoint slightly different than the FSAR nominal setpoint. However, the expected range of operation is not affected, and therefore, the operation of safety equipment is not affected. The existing tolerances meet accident analysis requirements. It is also desirable to maintain the existing tolerances and change the nominal setpoints slightly, rather than maintaining the nominal setpoints and changing the allowable tolerances.

All safeguards equipment with tolerances listed in the starting sequence in FSAR Section 8.2 still actuate within the FSAR-required tolerances. Therefore, there is no difference in operation for that equipment, and their accident mitigation function is performed. Diesel loading constraints (maintaining at least two seconds between starts of successive loads) and containment fan cooler time constraints (at least 35 seconds but no more than 50 seconds after safeguards actuation) are met. Therefore, for equipment with tolerances listed in the FSAR, the activity does not increase the consequences of an accident previously analyzed in the FSAR.

Where allowable tolerance bands are given in the FSAR, the changes to the nominal setpoints and associated tolerances meet the requirements of the FSAR tolerance bands. Therefore, for that equipment, the effective operating times of the safeguards are not affected. Where an allowable tolerance band is not listed in the FSAR (containment spray pumps), accident analysis and diesel loading criteria identified in the FSAR are met by the ORT-6 acceptance band, which bounds the setpoint associated tolerance. Therefore, the safety function of the containment spray pumps is not affected. (SER 95-059)

36. STPT 21.1, Protective Relay Setpoints for G-03, G-04 and New 1A-06 and 2A-06 Buses, Sheet 103 Revision 3. (Permanent)

As part of MR 91-116 to install new emergency diesel generators G-03 and G-04, setpoints for new protective relaying were installed during U2R20; Unit 2 Fall 1994 and U1R22, Unit 1 Spring 1995. The protective relay setpoints were for: G-03 and G-04 emergency diesel generators (EDGs); 1&2P-15B SI pump motor; 1&2X-14 transformer; 1&2X-06 transformer; 1&2A-06 new bus; and 4160 V degraded grid voltage and loss of voltage.

Summary of Safety Evaluation: A setpoint document change for these relay setpoints was previously approved. At that time a 10 CFR 50.59 screening was performed. During a subsequent root cause evaluation, it was determined a safety evaluation should be performed.

The basis for the setpoints is:

- 4160 V degraded grid voltage (DGV) relay setpoints and time delays were calculated to assure safety-related motors had adequate voltage to start and run continuously. The DGV relays trip the 1&2A-06 supply breaker. The settings are above the minimum value allowed by TS.
- 4160 V loss of voltage (LOV) relay setpoints and time delays detect a loss of offsite power, without tripping during normal plant voltage transients (such as RCP starts.) The relays trip the 1&2A-06 supply breaker, initiate start of the EDG, and permit closing the EDG output breaker. The setpoints are above the minimum value allowed by TS.

- The EDG voltage monitoring relay permits closing the EDG output breaker when the generator is up to voltage. The setpoint assures adequate voltage for safeguards loads and permits the EDG to accept load within 10 seconds.

- The EDG loss of field relay is set to detect loss of field due to short circuit or open circuit of the field supply, which could damage the generator. The EDG reverse power relay prevents motoring of the generator. Either relay trips the EDG output breaker if neither an automatic start signal, nor bus undervoltage is present.

- The EDG overpower relay and ground fault relay provide alarm only at the generator rated load.

- SI pump motor relays trip the motor for overloads or faults. While the relays used are different than those on the old A-06 buses (solid-state instead of electromechanical) the criteria for selecting the setpoints is the same and the level of protection remains the same.

  Transformer 1&2X-14 protection is provided by overcurrent relays on the transformer supply breakers for 1&2A-06. The relays coordinate with transformer low side breakers, and trip high and low side breakers. While the relays used are different than those on the old A-06 buses (solid-state instead of electromechanical) the criteria for selecting the setpoints is the same and the level of protection remains the same.

- Transformer 1&2X-06 protection is provided by overcurrent relays that trip the transformer supply breakers from 1&2A-06.

- Bus 1&2A-06 primary protection is provided by bus differential relays. The setpoints are selected so that the differential relays will not trip for faults external to bus 1&2A-06. Backup bus protection is provided by overcurrent relays on the breakers supplying 1&2A-06. The overcurrent relays are set to coordinate under the normal bus configuration, and with G-03 (G-04) EDG supplying the bus.

Protective relay setpoints protect equipment from damage, provide selective coordination, and preclude unnecessary tripping. DGV and LOV relay setpoints meet the requirements of TS. The other setpoints do not have TS requirements. (SER 95-069)

37. TS-44, Unit 1, In-place Testing of Main Steam Safety Valves, Revision 0. (New Procedure)
TS-45, Unit 2 In-Place Testing of Main Steam Safety Valves, Revision 0. (New Procedure)

The procedures fulfill the requirements for main steam safety valve surveillance testing required by ASME Code and TS Table 15.4.1-2, Item 12. Set pressure and leak rate test of the main steam safety valves is accomplished in-situ, using industry accepted lift assist technology. The actual testing is performed by a qualified National Board VR/NR Certificate holder, and does not affect the system or the intended functions of the components.

Summary of Safety Evaluation: The main steam safety valves have typically been tested during refueling outages by removing the valves and sending them to a qualified offsite testing facility. This process is very costly and expends valuable outage resources. Conventional testing of the valves offsite also causes greater wear on the components and increases the risk of accidental valve damage during shipping and handling.

System pressure (70-90% of valve set pressure) provides the motive force for testing. A hydraulic assist device is attached to the stem of each component (one at a time), and provides the additional force necessary to allow system pressure to overcome the valve spring load. The hydraulic assist equipment is fully instrumented. From the start of the test to the valve lift point, the applied load is linear with its slope representing the valve spring constant. At the time of valve lift, the slope of the applied load curve changes due to system pressure working on a larger exposed disc area. This change in slope is automatically sensed by the assist device computer, and the assist force is released allowing the valve to close under its own spring force.

During the entire evolution, valve disc lift is only approximately 0.031" for a fraction of a second before reseat occurs. The valve remains operable and valve response to an overpressure condition is not inhibited. Valve disc lift is minute, so no appreciable amount of steam is released to atmosphere. No visible affect on steam generator pressure or RCS coolant parameters (e.g., temperature, pressure, etc.) is expected. Other plants have also tested MSSVs in-situ without noticeably affecting steam generator pressure or RCS coolant parameters.

In the event a valve fails to reseat following the test lift, the assist device is equipped with a hydraulic ram (5,000 lbs closing force) that has the capability to reseat the valve disc if necessary. The probability of a valve to fail to reseat following lifting remains very low ($3 \times 10^{-3}$ failures per demand; reference INEL EGG-SSRE-8875 of February 1990), and is not affected by use of the lift-assist equipment. Nonetheless, the hydraulic ram is available for use to prevent continued steam release and valve disc damage due to steam cutting. Should this occur, the valve would be considered inoperable until repaired at an offsite facility and reinstalled.

The momentary lifting of each safety valve for testing purposes is considered operation of the valve within its design parameters. Operation of the lift-assist equipment and the lifting of the valve requires use of a dedicated operator as each main steam safety is part of the containment penetration boundary for its associated main steam line. (SERs 95-046, 95-104)

38. RAM 6.8, Multi-Assembly Sealed Basket (MSB) Radioactive Gas Discharge Permit, Revision 0. (New Procedure)

Prior to opening the MSB for unloading, the gas space surrounding the fuel must be sampled for gaseous radioactivity to determine whether or not a permit is needed to administer its controlled release via a monitored release point. A nitrogen purge is used to displace the radioactive gas from the MSB through existing piping designed for this purpose and exhausted to the drumming area vent stack.

Summary of Safety Evaluation: The situation that could arise from discharging radioactive MSB gas to a monitored gaseous effluent point such as the drumming area vent stack (DAVS) is an unexpected and uncontrolled release to the atmosphere of radioactive fission gases from a failed MSB or associated discharge lines. This situation is similar to the scenario described in FSAR 14.2.3, "Accidental Release - Waste Gas, Gas Decay Tank Rupture." In this situation, it is assumed that a gas decay tank with an inventory of 46,000 curies equivalent Xe-133 with no appreciable amounts of iodine and particulates has a gross tank or associated piping failure. A second situation described in FSAR 14.2.3, Accidental Release - Waste Gas, Volume Control Tank Rupture, would release 1,235 curies equivalent Xe-133. In both of these cases, 1% failed fuel is assumed. In the later situation, 25 mrem would be the total integrated dose at the site boundary control center (SBCC) using the conservative TID-14844 meteorology. The release from a

MSB or associated piping failure is calculated to be 15,100 curies of Kr-85. This curie concentration is based on a failure of 100% of the rods in the MSB and a release of 30% of the available fission gases. A release of 15,100 curies of Kr-85 from the MSB equates to a total SBCC dose of 1.4 mrem skin, 0.97 mrem whole body and would be less than current applicable 10 CFR 20 and 10 CFR 50 Appendix I limits. These doses are less than the dose resulting from the accident analyzed in the safety analysis report.

When the MSB gas space is to be discharged, a sample of the gas space is taken and analyzed per CAMP-607, "Multi-Assembly Sealed Basket Gas Sampling For Spent Fuel Unloading of Dry Storage Casks." CAMP-607 also issues a permit for gaseous discharge of the MSB based on the sample's isotopic results. If the gas is radioactive, a permit is issued to administer the release of the MSB gas. In the discharge flow path, at least two valves are manually opened: CRF-5, HX-298A&B outlet to vacuum drying system; and CDW-16, T-184 cask dewatering (CDW) moisture separator to the drumming area vent (DAV). An operator is assigned to manually shut CFR-5 when notified by the Control Room of an effluent high radioactivity signal from the DAV stack RMS monitor. Radioactive gas from the MSB is then routed to the DAV. The DAV is used as a dilution flow for the discharge flow path, and its radiation monitor (RE-221 or RE-325) is used to provide and maintain surveillance over the release. RAM 6.8 addresses a radioactivity release and provides guidelines to determine approval of releasing the gas.

The methodology and criteria required for a controlled release for other radioactive effluents are described in the FSAR and RETS. Controls equivalent to those used to discharge a gas decay tank (GDT) are used to control the discharge of potentially radioactive gas from the MSB since both discharges can be considered equivalent from a release viewpoint. In particular, both discharges are of a discrete volume of mainly long-lived radioactive gas isotopes via a controlled and monitored release.

Drumming Area Vent Stack radiation monitors RE-221 or RE-325 are used to maintain surveillance over the release. The sample activity of the volume of MSB radioactive gas to be discharged is determined by radiochemical analysis.

Instead of a flow meter to measure and regulate the discharge rate, an orifice plate, sized specifically for the designated discharge rate, is to be installed in the discharge line prior to the release to ensure the maximum flow specified on the MSB discharge permit is not exceeded. The orifice plate is sized according to MSB pressure and 200°F. An independent check is made and documented on the discharge permit verifying that the correct orifice plate is installed. The orifice plate rating is checked prior to installation.

Unrestricted area dose is calculated and documented on the MSB radioactive gas discharge permit. If the amount of radioactive gas to be discharged exceeds the limitations in TS 15.7.5.D.1 and 2, the MSB discharge is not discharged without further evaluation. However, this situation is unlikely since the worst case unrestricted dose calculated is 1.4 mrem. The annual average unrestricted dose of gaseous effluents is less than 1% of the annual RETS dose. Therefore, worst case release of radioactive gas for one MSB and the annual average gaseous release dose does not exceed the limits specified in RETS. (SER 95-092)

# MODIFICATIONS

The following modifications were installed as of the end of 1995:

1. MR 85-255*A (Common), Radiation Monitoring. MR 85-255*A replaces the software and memory boards of the microcomputers used in the data acquisition modules (DAMs), the special particulate, iodine and noble gas monitors (SPINGs) and the control terminals (CTs).

   Summary of Safety Evaluation: The MR improves the reliability of the RMS and makes the associated control functions more fail-safe. The modification addresses SOER 85-001.

   The memory boards provide the storage medium for the operating parameters of the RMS microcomputers including calibration constants, engineering units, various alarm setpoints, and channel file numbers. The existing memory boards (MEMI) have the potential for altering the operating parameters if power to the microcomputer is interrupted.

   The new memory boards (MEMII) are redesigned so that the operating parameters are not affected by a loss of power to the microcomputer. Except for the improved restoration of operating parameters following power interruptions, the MEMII boards are functionally identical to the MEMI boards and do not affect the operation or functions of the DAMs, SPINGs, or CTs.

   The replacement of the memory boards do not change the FSAR descriptions of the RMS, does not require a change to the TS and does not result in an unreviewed safety question.

   The RMS Eproms in the DAMs and SPINGs are also replaced. The Eproms contain the operating programs for the DAMs and SPINGs. The operating program has a priority system associated with the condition of the channel/detector. The priority arrangement is such that a fail high or fail low condition is of a higher priority than a high radiation condition. The existing design allows a detector to fail high or low without actuating the control functions associated with a high radiation alarm. This creates the potential, as described in SOER 85-001, where due to high radiation levels a detector fails high and the discharge path isolation valves do not automatically shut and an unmonitored release occurs.

   The new Eproms provide a new control bit which is actuated by a channel fail high, fail low or high radiation alarm signal. The control functions for each DAM and SPING channel are rewired to actuate by the new control bit instead of the high radiation alarm bit. Following this change, the new control bits operate the alarm and control function solid-state relays in the DAMs and SPINGs that were previously operated by the high radiation alarm bit. As a result, those RMS associated control and isolation functions and alarms previously actuated by a high radiation alarm signal are actuated by not only a high radiation alarm signal but also by a detector failed high or failed low signal. This eliminates the possibility of an inadvertent radioactive release through a release path with a failed detector.

   The control function for RE-101 and RE-235 shifts the Control Room ventilation to 100% recirculation to prevent contaminating the Control Room environment. The FSAR does not describe control function for the containment SPINGs.

The descriptions for the control functions are still correct following the modification because a high radiation condition initiates the required control function. The FSAR needs updating to reflect that a detector fail high or fail low condition also initiates the control function.

It is necessary to temporarily secure power to each DAM, SPING, and CT to install the modification. Installation procedures ensure the minimum requirements for radioactive effluent monitoring instrumentation operability found in TS 15.7.3 are no violated during the replacement of the memory boards.

The change does not pose an unreviewed safety question. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety is not increased. The change does not create the possibility for an accident or malfunction which has not been previously evaluated. The margin of safety as defined in the TS is not reduced. (SER 86-053-02)

2. MR 87-091 (Common), Radiation Monitoring. MR 87-091 provides for the functional replacement of the RMS smart terminals (CRT) in the Control Room and Technical Support Center. Display of information is available in the Control Room, the Technical Support Center and the Emergency Operations Facility. The radiological dose projection software is replaced by a more sophisticated program residing on the PPCS. This program is available via computer terminals in the Control Room, the Technical Support Center, and the Emergency Operations Facility.

   Summary of Safety Evaluation: FSAR Section 11.2.3, Radiation Monitoring System, is revised to reflect the new display configuration for RMS. The functional description in the FSAR of the CRT is revised to reflect the implementation on the PPCS.

   The modification has no affect on the sensing of the levels of radiation. The displays are equivalent or better than those previously used. Improvements are made in display of trends for some selected channels. The channels are those most indicative of changes in plant operating conditions. This should provide more timely response to radiation condition changes. The revised locations of the RMS monitor display in the Control Room enhances control operator use of it, as the PPCS is near the control boards. Real time trending of RMS channels should also provide early warning of abnormal conditions for those accidents listed in FSAR Chapter 14 as well as other conditions resulting in a change to the radiological conditions in the plant. (SER 88-045)

3. MR 88-136*C (Common), Computer. MR 88-136*C replaces existing Control Room operator consoles with consoles that address problems with workspace alarm screen access, procedure and supply storage, and fire loading. The new consoles have an additional terminal which is dedicated to alarm screen monitoring. The consoles are also seismically secured (Class 2).

   Summary of Safety Evaluation: The consoles are seismically mounted. The consoles fall under the category of Class II as defined in FSAR Appendix A: "Those structures and components which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity." The console mountings meet Seismic 2/1 design requirements. Therefore, the new consoles do not impact the operability of Class 1 equipment in the Control Room during a design basis earthquake (DBE). Since the consoles themselves are Seismic Class 2, they are not required to function after a design basis earthquake.

The consoles have an additional alarm screen monitor and associated instrumentation which affects the electrical load. A load analysis shows the instrument buses were running at 39.9% (1Y-03) and 43.6% (2Y-04) of rated load. The additional 1.3 amp load makes the loads 40.6% and 44.3%. The buses are still within rated limits. In addition, fuses are provided at the line input of the monitors to prevent a degradation of the instrument bus reliability in the event of a malfunction of the monitor.

Human factor concerns need to be incorporated in the design. FSAR Section 7.7 states that the control operators area of surveillance should be kept at a minimum. The modification adds a monitor to the current console configuration. However, this additional monitor actually reduces the operator's response time by providing a simultaneous display of a reactor monitoring screen and the alarm screen. The control operator currently swaps between the two screens on the PPCS monitor. The consoles are also designed in compliance with NUREG 0737 and NRC SER "Detailed Control Room Design Review (DCRDR)." These documents address the issue of human factors engineering.

Temporary Control Room configurations during installation period are considered. There is an interim period of 8-12 hours per console for the installation. the Unit 2 installation takes place immediately after the fuel is offloaded. Access to the PPCS is not necessary at this time for the control operators. A PPCS monitor is made available for use. The Unit 1 installation takes place immediately after the Unit 2 installation is completed. During the Unit 1 installation, the control operator does not have access to the SAS and PPCS terminals that normally reside on the console. However, they are able to use the C-03 display, the two monitors in the ASIP, the analog trend recorders in the ASIP, and the 2C-200 PPCS monitor.

The new consoles enhance operator control of the plant, improve the structural integrity of the operator consoles, and decrease fire loading in the Control Room. As a result, this change does not present an unreviewed safety question. (SER 89-122-01)

4. MR 89-133*A (Unit 1), 120 V Electrical. MR 89-133*A installs additional incandescent lighting and 120 V power receptacles in the Unit 1 containment loops. Additional support equipment include a power transformer and an electrical panelboard.

Summary of Safety Evaluation: The specified materials selected minimize, or eliminate if possible, the amount of aluminum added to the permanent containment inventory. Small amounts of aluminum may unavoidably be added to containment from the panel board bus bars and the luminaire lamp sockets; however, none of this aluminum is exposed. Additionally, two aluminum luminaries (weight = 15 1/2 pounds) added with MR E-142 are replaced, so there is a net decrease in both the total containment aluminum inventory and the amount of exposed aluminum in containment. Therefore, the permissible quantity of aluminum in containment is not exceeded by the installation.

The wire selected is acceptable measure it is tested in accordance with VW-1, the vertical wire flame test from UL-1581, conductors are fully enclosed within conduit, and this is not a Class 1E electrical installation.

Field painted surfaces are coated with post-design based accident (DBA) qualified paints. Some small items painted by the vendor may be supplied with a standard industrial-grade coating. In these cases, the items are installed away from the containment sump, and therefore pose no special safety concern. This determination is based upon the containment sump study.

Equipment installed is non-safety-related and non-QA scope. However, items installed could, during a seismic event, damage or disable equipment which perform a safety function and are in close proximity. Therefore, equipment supports are designed as QA-scope and meet the seismic Class 2 over 1 criteria. (SER 91-015)

5. MRs 89-165 (Unit 1)and 89-166 (Unit 2), Communications. The modifications change the Gai-tronics public address system in the reactor containments. The eight Gai-tronics horns installed in each containment provide the system with better sound coverage.

Summary of Safety Evaluation: Work is performed at the Gai-tronics isolation relay cabinets located outside of the containments. Therefore, containment entries are not necessary to perform the modifications. During portions of the installation, the Gai-tronics units inside containment are placed out of service for brief periods. However, since the work is performed on both units at power, no personnel are expected to be in the containments. If containment entry is required while the Gai-tronics units are inoperable, entering personnel are equipped with radios for emergency notification purposes. Following successful installation and acceptance testing, the Gai-tronics system is fully functional for use in the evacuation of either or both containments. Revised procedures reflect use of the Gai-tronics system in place of the containment evacuation alarm systems. The evacuation alarm systems remain operable, but are no longer used and will be disconnected and removed.

The systems provide important personnel safety and response coordination functions. The containment evacuation alarms are utilized to evacuate personnel in the event of a radiological event inside containment, while the Gai-tronics system provides the fire and evacuation alarms. The modifications improve evacuation of personnel from the containments and do not increase the probability of a failure of the Gai-tronics system. (SER 95-012)

6. MR 90-163 (Unit 1), and 90-164 (Unit 2), Radiation Shielding. The modification reduces personnel radiation exposure from the reactor vessel (RV) head and appendages. Radiation exposure is incurred from the RV head and appendages during each refueling outage. The shielding panels are stored on containment El 66' in stainless steel-lined storage boxes during plant operation.

Summary of Safety Evaluation: The addition of the new RV head shield structure does not affect safety-related systems or components, or equipment required to attain safe shutdown of the plant following a design basis accident (DBA). The permanent shielding ring is designed to withstand normal operating and safe shutdown loading conditions. This includes performance of a structural seismic review. Installation is performed when the plant is in cold shutdown and has no effect on the probability of an accident.

The consequences of an accident associated with the structure being modified are not explicitly discussed in the FSAR. This modification does not directly or indirectly initiate an accident. The design meets the design, material and construction standards applicable to the original structure. There is no potential for increased radiological consequences as a result of damage to a radiation barrier or the designated storage of the shielding during plant operation.

The integrity of the reactor coolant pressure boundary is not changed. The capability to shut down the reactor and maintain it in a safe shutdown condition is not affected. The RV head is not affected by the additional weight. The permanent shielding ring is designed to maintain its structural integrity during a safe shutdown earthquake. The permanent support structure does not compromise or impede the functionality of safety-related components/structures in the vicinity in the event of a safe shutdown of the reactor.

The RV head modification does not interface with the operation of the polar crane during the removal and installation phase. Shielding blankets are removed prior to a reactor vessel head lift due to the crane capacity and are reinstalled when the RV head is in the laydown area. The permanent addition of the RV head shielding ring contributes approximately 2000 lbs to the weight of the RV head; therefore affecting the total weight being lifted by the polar crane. This represents about 1% of the total weight of the RV head, which remains below the rated polar crane capacity of 200,000 lbs. (SER-94-042)

7. MR 90-013 (Unit 1), Safety Injection. The modification changes the limit switch contacts for 12 motor-operated valves (MOVs) and 2 air-operated valves (AOVs) that supply the safety injection (SI) spray ready status panel indications.

   Summary of Safety Evaluation: The modification changes the location of the wires that supply indication to the SI spray ready status panel to the correct contacts on each valve limit switch. The 12 MOVs have the required spare contacts available on its previously installed 4-rotor limit switches. The limit switches are adjusted and rewired. The two AOVs limit switches are also rewired to the available spare contacts. The corrections aid the operator by providing accurate indication on the SI spray ready status panel and providing better assurance of proper valve positioning for the associated systems.

   Valves 1SI-870A&B also have electrical interlocks with valves 1SI-871A&B being rewired. These interlocks, which prevent 1SI-871A&B (containment spray from RHR heat exchanger) from opening until 1SI-870A&B (containment spray from RWST) are fully shut, are currently supplied by auxiliary relay which also supply the current SI status light indication for 1SI-870A&B. With the removal of the SI status light indication for these auxiliary relays, the interlocks are hard wired into the circuitry using the valve limit switches and removing the auxiliary relays. This reduces the chance of failure of each interlock by removing one component which could possibly fail.

   The work was performed during U1R22 refueling outage as the valves were not required to be operational during this plant condition. (SER 95-004)

8. MR 90-086*D (Unit 1), Containment. The modification installs approximately 22' of handrails and one platform on El 76' and 100' of the Unit 1 steam generator shield walls. It also revises two ladders on the A steam generator shield wall.

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

Installation activities are not allowed if a potential exists for debris to enter the reactor coolant system (e.g., when the reactor vessel head is removed and the debris screen is not installed). This restriction applies only to installation activities on the sides of the steam generator shield walls that face the refueling cavity. These are the only locations where installation occurs that could introduce foreign material into the reactor coolant system. (SER 95-008)

9. MR 90-241*A (Common), Service Water. The modification adds flush connections to the service water (SW) supply headers at the suctions to the auxiliary feedwater pumps (AFPs). Problems with silt and sludge buildup were experienced at the low point of the SW riser just upstream of the SW supply motor-operated valves (MOVs) for the spent fuel pool. The modification removes the existing piping elbow and replaces it with a cross fitting.

Summary of Safety Evaluation: The changes do not affect the functionality of the auxiliary feedwater system. Ebasco Calculation 300024-EB and Sargeant and Lundy Calculation WE300024 Addendum A analyzed the additional weight of fittings and valves. The results show the piping and support changes remain within applicable criteria and Code stress allowables. The addition of the cross fitting in place of the existing elbow slightly affects the hydraulic characteristics of the system. However, the flushing capability removes silt deposits and makes the system more reliable. The installation is performed per applicable installation codes and procedures. Post-installation testing ensures that new components are capable of performing their intended function. Post-installation foreign material exclusion (FME) inspections ensure that the flow path is unobstructed.

The change adds potential leak paths to the SW line, however, the leakage would be very small and would not compromise the functionality of the auxiliary feedwater system. This change does not affect previous auxiliary feedwater pump room flooding. Administrative controls are utilized during flushing to ensure the safety and operability of nearby equipment is maintained, should a hose fail. The flood mitigating characteristics of the auxiliary feedwater pump room still allows time for operator action to isolate a failure prior to affecting other equipment. (SER 95-039)

10. MRs 90-262 (Unit 1) and 90-263 (Unit 2), Radiation Shielding. The modifications eliminate the temporary shielding and posting requirements at the containment annular gap during refueling operations. The modifications install permanent shielding by filling the air space between the fuel transfer tube (FTT) and the FTT penetration closure pipe with 1/4" diameter lead shot. Additionally, the existing local leak rate test connection for the FTT penetration is converted to a fill pipe/test connection to aid in the initial fill and allow periodic monitoring of the settlement of the lead shot.

Summary of Safety Evaluation: The configuration of the FTT penetration closure pipe establishes containment integrity with redundant barriers. During installation, the inner containment barrier established by the FTT penetration closure pipe is breached and containment integrity, as defined in TS 15.1.d., cannot be maintained. Therefore, per TS 15.3.6, installation requires that the reactor be in cold shutdown, fuel movement be suspended, and positive reactivity changes be restricted.

Installation does not impact the integrity of the outer containment barrier established by the FTT penetration closure pipe. Therefore, no additional or special actions are required to establish containment closure in accordance with containment closure checklist CL-1E.

The modifications have no impact on the ability of the FTT or the FTT penetration closure pipe to provide a leak-tight containment boundary. Having lead shot in direct contact with the stainless steel FTT and the carbon steel FTT penetration and containment liner does not induce corrosion or embrittlement of these components. The integrity of the FTT penetration closure pipe ensures that lead shot does not spill into the containment therefore impacting the operation of other equipment.

To minimize the potential for overexposure, a blind flange is used on the fill pipe to periodically monitor the settlement of lead shot within the FTT penetration closure pipe.

The FTT, the FTT penetration closure pipe, and the fill pipe are passive plant components which do not influence the performance of other components. (SER 95-022)

11. MR 91-116*F (Common), Fire Protection. MR 91-116*F installs a new 10" branch line and isolation valve in the north fire protection water supply header and the associated pipe and valves to provide fire protection water to Warehouse No. 4. It also installs a fire hydrant outside the security fence at the north gate.

    Summary of Safety Evaluation: During the installation, the section of the fire protection header that supplies Warehouse No. 3 and FH-24 is isolated. No work involving welding or burning is performed in Warehouse No. 3 during the installation. None of the systems listed in TS Table 15.3.14-1 are affected by the installation.

    The fire protection system is considered operable during the installation. The section of fire protection piping between PIV-83 and PIV-241 is isolated for approximately 5 days which is well within the TS limiting conditions for operation of 14 days. The hydraulic calculations for the sprinkler system in Warehouse No. 4 verify that the supply requirements for the sprinkler system in Warehouse No. 4 fall within the water supply capabilities of the fire protection water supply system. Therefore, this modification, nor its installation does not adversely impact plant safety as defined in TS FSAR or the Fire Protection Safety Evaluation Report and does not represent an unreviewed safety issue.

    Installation is in accordance with NFPA requirements for fire protection water system. It does not increase the system vulnerability or decrease the margin of safety as defined in the basis of the TS. (SER 93-025-02)

12. MR 91-116*K (Common), Radiation Monitoring. MR 91-116*K reinstalls radiation monitor 2RE-229/229B (service water effluent radiation monitor) in a new location. The modification reroutes the associated piping, conduit and cables and installs one new manual valve and flushing connection. The relocation accommodates the installation of new electrical equipment associated with the new emergency diesel generators.

    Summary of Safety Evaluation: Operational characteristics of the monitor are unaffected by this modification. Addition of the drain/flush valve and connection provides a location for periodic flushing of the tap which feeds the radiation monitor. This resolves silting problems which have previously affected this tap.

    The modification does not increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. Once installed, the radiation monitor provides the Control Room operators with the same radiation monitoring capabilities as before.

The modification does not create the possibility of an accident of a malfunction not previously evaluated in the FSAR. The only new system component installed is the drain/flush valve. This valve is of similar design as other valves installed. Any possible malfunction of the valve does not affect the monitor or the service water system since it is not directly in any flow path.

The modification does not reduce the margin of safety described in TS because the TS provides provision for the interim period when the monitor is inoperable. Once installed, all TS are met. (SER 93-025-08)

13. MR 91-116 (Common), Emergency Diesel Generators (EDGs). This modification installs two additional Class 1E EDGs. This evaluation addresses the final configuration.

Summary of Safety Evaluation: The new EDGs reduce the vulnerability of PBNP to a dual unit LCO shutdown based on EDG inoperability. The emergency ac power configuration using the new and existing EDGs provides the capability and capacity to allow one EDG to be inoperable in each train without being in an LCO on either unit if the remaining EDGs are placed in automatic mode to back up the operable EDGs. When more than two EDGs are available, the emergency ac power system is better than the old system.

The new EDGs and the reconfiguration of the emergency power system is in accordance with appropriate FSAR, design basis, and licensing basis requirements. TS were amended to account for this change in the emergency ac power configuration. (SER 93-025-24)

14. MR 91-116 (Common), Emergency Diesel Generators (EDGs), Final Configuration.

During modification scoping to change the W-2B G-01 vent fan power supply from 2B-32 to 1B-32, it was discovered the safety evaluation did not include evaluation of the G-01 day tank fuel oil fill valve, FO-3930. The valve is powered from the buses for which G-02 will be the normal emergency power supply.

Summary of Safety Evaluation: During the EDG final configuration, G-01 is normally aligned as the standby emergency power supply for 1A-05 and G-02 is normally aligned as the standby emergency power supply for 2A-05. The fuel oil fill valve for the G-01 day tank (FO-3930) is powered from 2B-32 from 2B-03/2A-05.

FO-3930 receives a signal to open automatically when the G-01 day tank level reaches 61%. The associated fuel oil transfer pump also starts, which automatically provides fuel oil to refill the day tank. The basis for TS 15.3.7 states, "The EDG fuel oil system is considered operable when ... 2) the EDG day tank for that EDG is operable and for G-01 and G-02 the associated motor-operated fill valve is operable. ... However, the fuel oil transfer system is allowed to be out-of-service for four hours for G-01 and G-02 due to a combined four-hour supply of fuel oil in the diesel base and day tanks which do not require a fuel oil transfer pump for flow to the associated EDG."

If emergency ac power is inoperable for 2A-05, FO-3930 would not have emergency power, and would not be able to operate automatically if offsite power is lost

Provisions for maintaining EDG operability, without automatic make-up to that EDG day tank were established. Appropriate compensatory actions are provided to give operators guidance to maintain the operability of G-01 when emergency ac power is inoperable for 2A-05. Otherwise, if the compensatory actions are not taken, G-01 must be declared inoperable, in accordance with the Basis for TS 15.3.7 until emergency ac power is restored to 2A-05. (SER 93-025-25)

15. MRs 91-116*R and T Common), Emergency Diesel Generators (EDGs). The assign packages install additional cables and extended tray covers on cable trays within the cable spreading room. The trays contain safety-related cables. The extended tray covers are installed to accommodate the addition of new cables required for the EDG project which results in some of the trays being filled above their existing side rails. This change (which requires removal of the existing tray covers and fire blankets, installation of the new cables, installation of the new extended tray covers and fire blankets, and penetration of existing fire barriers) is done while Unit 1 and Unit 2 are at power.

Summary of Safety Evaluation: The modifications comply with FSAR requirements with the exception of FSAR 7.2.1 which states, "When loading the cables into the trays, the height of cable bundles is maintained equal to, or below, the height of the tray." The seismic and ampacity implications of installing cables above the siderails were analyzed and demonstrated to be acceptable. An FSAR update will incorporate this exception.

The seismic capability of the trays and their supports were evaluated in accordance with original plant cable tray design criteria or SQUG (Seismic Qualification Utilities Group) criteria. The structural capacity of the existing or modified tray and existing supports is greater than the applied loading (see UE&C Calculation 6704.001-C-075).

The cable is fire rated in accordance with IEEE-383. Fire blankets are reinstalled as part of the final installation. Temporary fire blankets are used during installation.

Tray 1EM01 is 39.8% full, and therefore acceptable. Trays 2EK02 and 2AQ01 are greater than 40% so additional cables are routed through the floor in the same opening as the tray. However, these cables are routed as a series of cable bundles independent of the tray maintaining the required clearances (0.75") between the bundles and the tray and between the bundles and the floor opening. This approach results in a configuration that is equivalent to two independent penetrations. The 0.75" clearance is in accordance with "ANI/MAERP RA Guidelines for Fire Stop and Wrap Systems at Nuclear Facilities," Revision 0, Appendix A, Paragraph A-4. (SER 93-025-17)

16. MR 91-116 (Common), EDGs. The evaluation addressed Phase 3B of the EDG additions. It is an interim configuration of the electrical distribution and fuel oil systems established during U1R22.

Summary of Safety Evaluation: The old 1A-06 switchgear is replaced. This is an improvement over the old arrangement of the 4160 V emergency power buses. The lack of sufficient separation of 4160 V in the vital switchgear room was a major concern for fire scenarios in that area. Moving one train of 4160 V switchgear out of this room is a major improvement in separation. The new Train B equipment used is equivalent or better than the old emergency power equipment.

G-03 EDG provides standby emergency ac power for only 1A-06 and G-04 provides standby emergency ac power for 2A-06 and backup capability for 1A-06. G-02 is placed out of service for the modification to become a Train A EDG.

The Train B service water and auxiliary feedwater pump motors are supplied by the following 480 V safeguards bus arrangement: 1B-04 powers service water pump P-32C; and 2B-04 powers electrical auxiliary feedwater pump P-38B and service water pumps P-32D and P-32E.

The loss of normal power supply or standby emergency power supply for 2B-04 results in a loss of redundancy for both units because of the shared equipment power supply arrangement. Therefore, the LCO for the loss of normal power supply or standby emergency power supply for either bus must apply to both units. The loss of normal power supply or standby emergency power supply for 1B-04 does not result in an LCO for the unaffected unit, because sufficient redundancy remains (i.e., both electric auxiliary feedwater pumps and at least two service water pumps in each train). The applicability of the LCO to one or both units is determined by which bus has lost normal or standby emergency power capability. TS establishes the requirements for operability of the emergency power system.

Selection of the EDG that will automatically energize the 1A-06 safeguards bus upon an undervoltage condition is made by placing the associated EDG (G-03) output breaker control switch in the "auto" position. G-04 does not automatically energize 1A-06 because it has its 1A-06 output breaker control switch key locked in the pull-out position. The control switch for the G-03 output breaker must be in pull-out when both the output breakers from the G-04 are in "auto" to reduce the possibility of accidental paralleling of two EDGs. Also, the output breakers have interlocks that prevent closure of the G-04 output breaker if the output breaker from G-03 is closed to that bus. The EDG system could still perform its safety function even if accidental parallel operation of the EDGs occurred, because only one train would be affected. The failure of one train of emergency power has been previously evaluated in the FSAR.

New breaker A52-54 is connected to 1A-04 to provide protection against a fault on the new cable from 1A-04 to the new 1A-06.

This breaker provides additional protection and reliability of the 1A-04 bus. The breaker addition allows 1A-04 to remain operable if a fault occurs in the new cable from 1A-04 to the new 1A-06 in the new EDG building. The additional breaker is in the non-safety-related portion of the electrical distribution system. The reliability of the normal emergency power supply 1A-04 is not significantly reduced, because the new breaker is expected to be as reliable as the existing breakers in this portion of the electrical distribution systems, which are very reliable. Even if the new breaker would spuriously open, causing an interruption of the normal emergency power supply to 1A-06, the new EDG G-03 would be expected to start and reenergize the 1A-06 bus.

The new EDG building, and the fuel oil and cable runs are designed and constructed with appropriate protection and integrity to minimize the risk of failure because of exposure to hazards outside the plant. G-03 and G-04 are radiator cooled. The radiator cooling is different than the cooling of the existing EDGs. The existing EDGs are cooled by service water. The design and construction of the radiator cooling system is such that it is not more susceptible to failure than the cooling system for the existing EDGs. Also, this makes the emergency ac power system less susceptible to common mode failure because of the diverse means of cooling the existing and new EDGs.

For the time being, the existing fuel oil system continues to provide fuel oil for G-01. The new fuel oil tank T-175B, fuel oil transfer pumps P-206B (powered by G-03) and P-207B (powered by G-04), and the associated fuel supply system for G-03 and G-04 are placed in-service.

Currently, there is not a design basis evaluation for the loss of power after an accident that initiates safety injection (SI). Accidents that initiate SI analyzed in the FSAR assume that the loss of power occurs concurrent with the accident/reactor trip, or not at all. An evaluation of the loss of offsite power for 2 seconds, 50 seconds after the accident is contained within QCR 94-003.

The Unit 2 SI situation is not a significant change from the original design because required safety equipment would respond except service water pump P-032C would be delayed 10 seconds instead of 2 seconds if G-03 had an anticipatory start for Unit 2 SI. This would result in a total time delay of 25 seconds for P-032C, which is less than the time delay for restart of the last Train B service water pump P-032E, 27 seconds. Therefore, G-03 does not need a Unit 2 SI anticipatory start.

During this interim configuration (Phase 3B), emergency ac power capability is provided by three EDGs: G-01 to 1A-05 and/or 2A-05, G-03 to 1A-06, and G-04 to 2A-06, with backup capability from G-04 to 1A-06. The installation of G-03 and G-04 and the reconfiguration of the emergency power system is in accordance with the appropriate FSAR, design basis, and licensing basis requirements. (SER 95-001)

17. MR 91-116*AF (Common), EDGs. The modification completed the installation of the main feed to panel D-40 from bus D-02. The main feed to D-40 is not energized during the installation. The main supply cable is pulled, trained, labeled, lugged, connected and tested. Energization of D-40 through the normal supply switch D72-203 occurs via MR 91-116*X during U1R22.

Summary of Safety Evaluation: The FSAR does not indicate that the existing 125 Vdc system causes or affects the probability of an accident evaluated. The modification design meets or exceeds the existing system design requirements. The new cabling is qualified to meet the requirements of IEEE 383-1984 flame testing. The panel and associated switches are of a type not previously used; however, they are a proven commodity as an industrial component and are an improvement of the old devices. Precautions are taken while working on or near energized equipment.

Energization of D-40 from D-02 is isolated during this work. Switch D72-203 at D-02 and D72-40-M at D-40 remain open and isolated. Energization of D-40 through normal supply switch D72-203 occurs when MR 91-116*X is installed. The modification does not change the operating configuration of the dc system, nor does it increase the possibility of creating an unanalyzed condition in the 125 Vdc system. (SER 95-005)

18. MR 91-119*AE (Common), EDGs. The modification cuts a mounting hole in the C-002 front bench section for new sync switch SS/1A52-77; labels the control switch and cubicle as "not available for use" for breaker 1A52-61; disconnects the field cable for the control switch and sync switch for breaker 1A52-61; removed the control switch and sync switch for breaker 1A52-61; wires the 4160 V breaker control switch 1/1A52-84; terminates the cables in panels C-001 and C-002 on spare terminal points; and prepares internal wires in panel C-002 for future termination work.

Summary of Safety Evaluation: The only function in the FSAR changed by this activity is the removal of the 1A-05/06 bus tie capability (breaker 1A52-61). FSAR Section 8.2.1 states, "The bus tie breakers (1A52-61, 2A52-72, 1B52-16C and 2B52-40C) are supplied to facilitate maintenance of the normal supplies to the respective buses. Breakers 1A52-61 and 2A52-72 have been physically removed from the cubicle and placed in storage outside the safe shutdown area." This change removes the capability to cross-tie the 1A-05 and 1A-06 buses. However, this capability is only used for maintenance and is not required for normal operation or to mitigate an accident.

The abandonment of the 1A-05/1A-06 bus-tie feature does not adversely affect the plant's capability to meet TS and any condition or the assumptions that might have been made in the accident analysis.

Before the termination of the pre-outage cables in the control board, the terminal points are first checked to verify that no cables are terminated on the external side of the terminal blocks. The terminal block sliders are then opened to prevent interaction between the new cables and existing internal wiring. The conductors are then landed at the respective terminal points. The respective sliders are closed later and tested and the circuits are put into service. In addition, the control board work requires cutting to install a new control switch. Installation procedures prevent the degradation of the existing structures and components (e.g., vibration, affect on other components, exclusion of foreign materials, etc.) resulting from the installation. The installation of the cables and components are designed to meet the applicable design material, and construction requirements of the plant system and components. Following removal of sync switch for breaker 1A52-61, the affected sync circuits are checked to verify proper operation. (SER 95-019)

19. MR 91-116*AH (Common), EDGs. The modification changes a design error identified on drawings related to the wiring of the G-04 EDG differential relay. The Model SA-1 generator differential relay vendor testing was limited to the output of the vendor devices and the interface with field installed cable and equipment. Therefore the operation of the generator differential relay was not tested. A review of the wiring diagrams and relay manufacturer's literature showed that the vendor design incorrectly wiring the relay power supply across terminals (1) and (10). The actual 125 Vdc supply to the relay shall be applied across terminal (1) and (2). The design also incorrectly assumes the trip signal output of the relay is found at terminal (2). This signal is actually at terminal (10). This discrepancy was documented via the condition reporting system. The scope includes the swapping of leads on terminals (2) and (10) on the differential relay and testing the new wiring configuration to verify the relay functions properly.

Summary of Safety Evaluation: The EDG differential relay operates for any fault within the protective zone of the differential current transformers. The relay operates because of unbalanced currents on either side of the relay coil sums. The differential relay contact is used to energize an auxiliary relay. This relay in turn actuates an alarm to indicate the differential condition, trips the generator output breaker and initiates an emergency shutdown of the EDG. This protects the generator from a ground fault or a phase-to-phase ground. The original design and subsequent installation of the differential relay as part of MR 91-116*W intended that the relay perform in this manner. The relay does not function to protect the generator from a fault current. The error, however, does not affect the ability of the engine to start and load in the event of an emergency. The modification changes the relay wiring so that it performs its intended function. The modification shall be performed when G-04 is taken out of service to facilitate the G-03 tie-ins during the installation of MR 91-116*X. (SER 95-020)

20. MR 91-116*AJ (Common), Emergency Diesel Generators (EDGs). The design package reconfigures G-02 EDG controls in main control board C-02 to match those of new EDGs G-03 and G-04 and retrains G-02 from Train B to Train A. It removes existing G-02 controls in C-02, cutting out the area where the controls were mounted, and installs and rewires a new subpanel on C-02 containing the reconfigured G-02 controls.

Summary of Safety Evaluation: Activities occur while G-02 EDG is out of service. The subpanel installation is limited to main control board C-02. Panel C-02 also contains components for the G05

Cutting is done on panel C-02 front to facilitate the installation of the new subpanel. Similar cutting was previously performed without a safety concern. Mounting of the cutting equipment is seismically verified as adequate.

The installed subpanel uses existing G-02 EDG components or new Class 1E components where necessary. New internal wiring is SIS-qualified to IEEE-383 and routed using Train A wireways when appropriate to meet train separation requirements. A seismic adequacy verification of component, subpanel, and conduit mounting is performed. Functional testing of G-02 EDG controls are completed as part of MR 91-116*Y2 to ensure proper operation. (SER 95-061)

Summary of Safety Evaluation: This safety evaluation revision addresses additional considerations because of the presence of hydraulic lines in the Control Room while the control board cutting tool is used. Hydraulic lines are routed into the Control Room as part of the cutting process. Although the likelihood of a hydraulic line rupture is very small, the lines are sleeved in plastic to prevent splattering of fluid. A kill switch for the compressor is accessible to deplete the hydraulic pressure in the lines to minimize affects of a line leak. In addition, the hydraulic fluid poses no flammability concerns and operation of the hydraulic system is well within its maximum rating. Fluid lines are routed so no risk is posed to internal control board wiring. (SER 95-061-01)

21. MR 91-116*Y (Common), Emergency Diesel Generators (EDGs). Phase 3B disables the output from G-02 EDG to 1A-06. G-02 EDG does not provide standby emergency power for any bus during Phase 3B. While G-02 EDG is out of service the existing G-02 is retrained as a Train A EDG. The new Train A fuel oil transfer system is also placed in service for G-01 and G-02 EDGs. After G-02 is retrained and tested, it is placed in service to 2A-05 as the normal standby emergency supply. The modification also connects G-03 EDG to 2A-06 Train B as the alternate standby emergency supply.

Summary of Safety Evaluation: The design of the retraining and the fuel oil transfer system is performed to assure the new systems meet or exceed the existing system design requirements. Evaluations of the new power and control cable routings assure that applicable separation requirements are met. Piping for the new fuel oil system is installed per MR 91-116*E and *I. Piping stress and hydraulic calculations evaluated the system configuration. This system utilizes the new fuel oil storage tanks and transfer pumps located in the new EDG building to supply fuel oil to G-01 and G-02 EDGs.

The system provides a greater fuel inventory and a Seismic Class 1 storage system. The new fuel oil transfer system configuration may, however, allow a siphon to be established between the storage tank and the G-01 or G-02 EDG day tank if MOV-3931 should fail open. If the storage tank is at its maximum capacity, approximately 800 gallons of oil would spill from the tank vents. The existing vents on day tanks are extended to prevent a fuel oil spill in the event of an MOV failure. Spilling 800 gallons of fuel oil is not a significant loss of inventory and does not create a nuclear safety concern. The retraining of G-02 requires that the power supply for W-12C be switched from 1B-41 and 2B-32. The power supply for W-12D is switched from 2B42 to 2B-32. The power is changed from D-02 to D-03. Evaluations verify the additional loads placed on D-03, 2Y-06, 1B-32 and 2B-32 for the new fuel oil system and the G-02 auxiliaries are acceptable. Separate safety evaluations are performed for the energization and testing of the retrained equipment and systems.

To ensure that the work activities do not affect operating plant equipment or systems, connections are isolated from operating plant equipment by isolating the associated disconnect switches, circuit breakers and sliders open. The isolation tags are not cleared and isolation devices not closed until approved test procedures are issued. Equipment is not tested or placed in service while performing IWP 91-116*Y1. (SER 95-062)

22. MR 91-116 (Common), Emergency Diesel Generators (EDGs). Phase 3C an interim configuration of the electrical distribution system, is established during U2R21. Connection to existing systems is evaluated separate from the tie-in procedures. In this configuration G-03 continues to provide standby emergency ac power for 1A-06 and becomes a backup for 2A-06; the existing old 2A-06 bus becomes part of the 2A-05 bus and is included in the differential relaying protection circuitry for 2A-05; G-02 is changed to a Train A EDG and provides standby emergency ac power for 2A-05 during this phase; and G01 becomes a backup standby emergency power supply for 2A-05.

Summary of Safety Evaluation: The new EDGs are installed to reduce the vulnerability of PBNP to a dual unit LCO shutdown based on EDG inoperability. The emergency ac power configuration using the new and existing EDGs provide the capability and capacity to allow one EDG to be inoperable in each train without being in an LCO on either unit if the remaining EDGs are placed in an automatic mode to backup the inoperable EDGs. When more than two EDGs are available in the final configuration, the emergency ac power system is better than the old system.

During Phase 3C, emergency ac power capability is normally provided by four EDGs: G-01 to 1A-05 with backup capability to 2A-05; G-02 to 2A-05; G-03 to 1A-06 with backup capability to 2A-06; and G-04 to 2A-06 with backup capability from G-04 to 1A-06.

The independency of Trains A and B is maintained. The capacity of the emergency power system is not affected by this phase of the modification. The EDG capacity evaluation shows that the configuration provides sufficient margin on the EDGs. TS 15.4.6 requires monthly load testing of each EDG and emergency load testing during each refueling outage. The EDGs continue to be tested in this manner. G-01 continues to be tested to 1A-05 and when necessary to 2A-05 during performance of TS-81. G-03 is tested to 1A-06 and when necessary to 2A-06. G-04 continues to be tested to 2A-06 and when necessary to 1A-06. G-02 is tested to 2A-05. (SER 95-073)

23. MR 91-116 (Common), EDGs. This revision includes evaluation of the G-01 fuel oil tank fill valve, FO-3930, being powered from 2A-05 during Phase 3C.

Summary of Safety Evaluation: During Phase 3C, G-01 EDG is normally aligned as the standby emergency power supply for 1A-05 and G-02 is normally aligned as the standby emergency power supply for 2A-05. The fuel oil fill valve for the G-01 day tank (FO-3930) is powered from 2B-32 from 2B-03/2A-05.

The fuel oil fill valve for the G-01 EDG day tank receives a signal to open automatically when the G-01 day tank level reaches 61%. The associated fuel oil transfer pump also starts, which automatically provides fuel oil to refill the day tank. The Basis for TS 15.3.7 states, "The EDG fuel oil system is considered operable when . . . 2) the EDG day tank for that EDG is operable and for G-01 and G-02 the associated motor-operated fill valve is operable. . . . However, the fuel oil transfer system is allowed to be out-of-service for four hours for G-01 and G-02 due to a combined four-hour supply of fuel oil in the diesel base and day tanks which do not require a fuel oil transfer pump for flow to the associated EDG."

If emergency ac power is inoperable for 2A-05, the fill valve to the G-01 day tank would not have emergency power, and hence would not be able to operate automatically if offsite power is lost. G-01 can be considered operable in this situation, because provisions for maintaining EDG operability, without automatic make-up to that EDG day tank were established. Appropriate compensatory actions were established to provide the operators guidance to maintain the operability of G-01 when emergency ac power is inoperable for 2A-05. Otherwise, if the compensatory actions are not taken, G-01 must be declared inoperable, in accordance with the Basis for TS 15.3.7, until emergency ac power is restored to 2A-05. (SER 95-073-01)

24. MR 91-135*B (Unit 2), Main Steam. The modification replaces 2MS-2016, the Unit 2 A steam dump valve.

Summary of Safety Evaluation: The modification was installed during U2R21. The new 600 lb globe valve is similar to the old except for the trim design. The new valve balances the pressure differential on the valve plug to eliminate valve failures to open. The cage is designed to give the valve a better throttling characteristic in the 10-20% flow range for improved controllability. The new valve trim configuration reduces steam cutting and prolongs the life of the valve trim. The valve is installed and tested for operation and leak tightness.

The atmospheric steam dump lines are relied upon following a steam generator tube rupture coincident with a loss of ac power to cool down the reactor coolant system to RHR entry conditions. FSAR 10.2 states that the atmospheric power-operated relief valve is available for removal of sensible and core decay heat to atmosphere. This valve is automatically controlled by pressure or may manually be operated from the main control board and has a total capability of 666,400 lb/hr steam flow. The new valve has operating range and capacities identical to the original valve. Valve 2MS-2015 was replaced during U2R20 with successful operation noted since replacement.

An atmospheric steam dump line is considered operable if it is capable of providing the controlled relief of main steam flow necessary to perform the RCS cooldown. Isolating an atmospheric steam dump line does not render it inoperable if the line can be unisolated and the RCS still cooled down to RHR entry conditions, through local or remote operation, within the time period required by the applicable FSAR accident analyses. The valve replacement reduces the frequency and duration of having a steam dump path out of service. (SER 95-101)

25. MR 91-159 (Unit 2), Component Cooling Water. The component cooling (CC) water pump modification retains an operable spare rotating assembly for the CC pumps. The manufacturer was unable to repair the current spare assembly so a new rotating assembly was required. In addition, the CC pumps have experienced excessive mechanical seal and shaft bearing failures. This modification eliminates the seal and bearing failures while providing an operable spare rotating assembly.

Summary of Safety Evaluation: The bearing type is changed to limit the end play of the rotating assembly. The present configuration allows the rotating assembly to hunt continuously in the axial direction. This movement causes excessive wear on the mechanical seals. The original arrangement was a single row bearing while the new bearing is a double row bearing. The new bearing is superior to the old bearing on both axial and radial load carrying capability. The increased load capability is used to limit end play. The new bearing is wider than the old bearing and therefore requires lengthening the shaft by 15/16" and changing the thrust bearing housing to accommodate it.

The impeller material is upgraded from ASTM B143 Alloy 992 bronze to ASTM A487 stainless steel. The new impeller material has superior erosion, corrosion and abrasive wear resistance. In addition, the new material allows the impeller to be weld repaired if required. The seismic qualification of the CC pump is not affected by the new impeller. The new impeller is 15 pounds lighter than the old impeller.

The casing ring material is upgraded from the cast iron to ASIS 410 stainless steel. The stainless steel is superior to the cast iron in this application. The upgraded material requires the clearance between the impeller and casing rings to be increased to prevent the possibility of galling. Pump performance is not affected by this change.

The mechanical seals are upgraded from a John Crane Type 1 mechanical seal to a Chesterton Type 442 split mechanical seal. This change aids in insatiability of the seals, minimizing the down time of the pump. The new seals do not change the performance characteristics of the CC pump.

This modification does not affect the operability of the CC pump. The design characteristics of the pump remains unchanged. The modification results in a negligible change in weight to the CC pump and therefore has negligible affect on the seismic qualifications of the pump. (SER 95-117)

26. MR 92-053 (Unit 1), 125 Vdc Electrical. The modification corrects a potential power supply train separation problem in the circuits which start the 1P-29 turbine-driven auxiliary feedwater pump (TDAFP) on an undervoltage of the 1A-01 and 1A-02 buses. It also incorporates the replacement of timing relays 1-62A and 1-62B, which open 1P-29 TDAFP steam-supply MOVs 1MS-2019 and 1MS-2020 on 1A-01 and 1A-02 undervoltage, with more accurate and reliable electronic relays.

Summary of Safety Evaluation: The non-safety-related circuits for starting 1P-29 on 1A-01/1A-02 undervoltage and the safety-related circuits for initiating auxiliary feedwater flow on a low-low steam generator level were supplied from the same safeguards 125 Vdc sourced in main control board 1C-03. No electrical isolation devices (e.g., fuses or circuit breakers) were installed to segregate the non-safety-related circuits from the safety-related circuits supplied from the same power supplies. Furthermore, Trains A and B power supply cables terminate to adjacent terminal positions in the 1A-01 and 1A-02 switchgear cubicles, and Trains A and B wires were routed in the same wire bundles. The possibility therefore existed that a single fault could incapacitate not only both of the non-safety-related circuits for starting 1P-29 TDAFP on 1A-01 and 1A-02 undervoltage, but also both trains of safety-related circuitry for initiating auxiliary feedwater flow on a low-low steam generator level. This configuration is a violation of both our single failure criterion and the guidelines set forth in DG-E07, "Separation of Electrical Circuits."

The field cables for the old safeguards power supplies to its circuits are disconnected at both 1C-03 and the switchgear cubicles, solving the separation problem. The power source for the former Train A start circuit is the 1A-01 switchgear source, which is ultimately supplied by the Train A D01 dc bus. Powering the undervoltage start circuits from these sources maintains the old train configuration of the power supplies.

To protect the 1A-01 and 1A-02 switchgear power supplies in the event a fault on one or both of the undervoltage AFP start circuits, these circuits are powered from existing fuses located in the switchgear cubicles. These fuses presently supply the 1A-01/1A-02 undervoltage relays and their associated auxiliary relays. The configuration of the circuits which start 1P-29 TDAFP 1A-01 and 1A-02 undervoltage is such that they are disabled by a loss of power to the 1-271X1 or 1-272X1 auxiliary relays, since the normally open contacts on these relays would be unable to close if power to the relay coils is lost. Powering both the auxiliary relays and the 1P-29 TDAFP start circuits does not result in the fuses being overloaded. Furthermore, melt time curves for the fuses indicate it provides adequate coordination with its upstream supply breakers in the event of a short-circuit fault in the circuits it supplies.

The 1-62A and 1-62B time-delay relays that start the 1P-29 TDAFP on the 1A-01/1A-02 undervoltage (by opening 1P-29 steam supply valves 1MS-2019 and 1MS-2020) are replaced. The old Agastat Type 2412PC relays are replaced because its qualified service life times are exceeded. The relays have demonstrated a significant drift in their delay times. The new Agastat Type ETR14 relays, feature a solid-state timer which allows greater accuracy and reliability.

Installation occurs with Unit 1 in a cold or refueling shutdown condition and the 1P-29 TDAFP isolated out-of-service. During the installation, the automatic operation of 1MS-2019 and 1MS-2020 are disabled so work may safety take place on the associated start circuits. Since 1P-29 TDAFP is isolated out-of-service, automatic operation of these valves is not required. Power to the safety-related circuits for starting 1P-29, P-38A and P-38B on a low-low level in either or both of the Unit 1 SGs is isolated briefly while the existing power supplies for the 1P-29 start circuits are disconnected at 1C-03. However, with Unit 1 in cold shutdown, the circuits are not required to be operational, so power may be removed without consequence. Installation does not jeopardize the ability of P-38A and P-38B motor-driven auxiliary feedwater pumps (MDAFP) and 2P-29 TDAFP to start and supply feedwater to the Unit 2 SGs on either a 2A-01/2A-02 undervoltage signal or a low-low steam generator water level signal for Unit 2. (SER 95-013)

27. MR 92-054 (Unit 2), 125 Vdc Electrical. The modification corrects a potential power supply train separation problem in the circuits which start the 2P-29 turbine-driven auxiliary feedwater pump (TDAFP) on an undervoltage of the 2A-01 and 2A-02 buses. This separation problem was corrected was corrected by changing the power supplies for these circuits, which are not considered safety-related, from safeguards to non-safeguards dc power. Replacement of timing relays 2-62A and 2-62B with more accurate and reliable electronic relays. These relays open 2P-29 steam-supply MOVs 2MS-2019 and 2MS-2020 on 2A-01 and 2A-02 undervoltage.

Summary of Safety Evaluation: The installation takes place with Unit 2 in cold or refueling shutdown and 2P-29 TDAFP out-of-service. During the installation, operation of the 2P-29 steam supply MOVs (2MS-2019 and 2MS-2020) is disabled so work may safely take place on the associated start circuits. Since 2P-29 is out-of-service, operation of these valves is not required. Power to the safety-related circuits for starting 2P-29, P-38A, and P-38B on a low-low level in either or both of the Unit 2 steam generators is isolated briefly while the existing power supplies for the 2P-29 start circuits are disconnected at 2C-03. However, with Unit 2 in cold shutdown, these circuits are not required to be operational, so power may be removed without consequence. Installation activities at do not jeopardize the ability of the motor-driven pumps (P-38A and P-38B) and the Unit 1 turbine-driven pump (1P-29) to start and supply feedwater to the Unit 1 steam generators on either a 1A-01/1A-02 undervoltage signal or a low-low steam generator water level signal for Unit 1. The operability of affected circuits is verified prior to Unit 2 leaving cold shutdown.

Neither the configuration changes nor the interim conditions associated with the modification have the potential to initiate an accident, previously evaluated in the FSAR or otherwise. The configuration changes do not result in functional changes to circuits, and do not increase the probability of a malfunction in circuits or equipment, including those necessary to mitigate the consequences of an accident or safety-related equipment malfunction. Thus, the configuration changes do not increase the consequences of accidents or equipment malfunctions, and do not create the possibility of new malfunctions. The installation takes place with the plant configured such that none of the circuits affected by the installation required to operate to mitigate the consequences of an accident or safety-related equipment malfunction. The modification does not alter the functionality of circuits, and does not introduce new failure mechanisms to circuits or safety-related equipment. (SER 95-097)

28. MR 92-120 (Common), Independent Spent Fuel Storage Installation (ISFSI). The evaluation reviews the ventilated storage cask (VSC) system. The VSC system is a dry storage system utilizing a concrete storage cask, and a steel, seal-welded multi-assembly sealed basket (MSB) to store irradiated nuclear fuel.

The VSC system includes a ventilated concrete cask (VCC); 24-assembly multi-assembly sealed basket (MSB); MSB transfer cask (MTC); vacuum drying and helium (He) backfill system with a helium sniffer for leak detection; and an engineered cask transporter.

Summary of Safety Evaluation: The VSC is licensed under 10 CFR 72 Subpart K "General Licenses for Storage of Spent Fuel at Power Reactor Sites." The VSC system was reviewed and approved by the NRC and has received a NRC SER and a Certificate of Compliance. The safety evaluation complies with Subpart K Section 72.212 (b)(4).

Existing fuel handling equipment is used to handle the fuel assemblies in the spent fuel pool (SFP). Procedures address loading of the spent fuel into the MSB rather than an SFP rack. Otherwise, fuel handling is performed in the usual manner. Fuel handling personnel are trained prior to actual fuel movement. The FSAR recognizes that fuel is loaded into shipping casks for disposal offsite. Fuel loading for offsite shipments is similar to the basic steps of the MSB/VCC loading. Therefore, given the use of existing fuel handling equipment and procedural and administrative controls, there is no increase in probability of a fuel handling accident as described in the FSAR.

Since the fuel moved during the loading of the MSB is a minimum of 5 years old, has a maximum initial enrichment of 4.2 wt% U-235, and only one assembly is moved into the MSB at a time, the consequences of a fuel handling accident while moving that fuel assembly are bounded by those analyzed in the FSAR and subsequent NRC Safety Evaluation Reports. Therefore, there is no increase in consequences of an accident during loading of the MSB.

There is no equipment important to safety in the vicinity of the SFP, decontamination area and north truck access. The transfer route of the cask from the SFP to the decontamination area and then to the truck access area of the primary auxiliary building follows an approved heavy load path identified in accordance with NUREG-0612. (SER 94-041)

29. MR 92-120*D1 (Common), ISFSI. The modification includes only the mechanical portion of the installation and operation of the vacuum drying system (VDS) and the cask reflooding system (CRS) and its potential effects on existing systems, structures and components affecting the 10 CFR 50 license. The capabilities of the system to perform their functions relative to dry storage are addressed in 10 CFR 72.212 (b)(2) evaluation.

The vacuum drying system includes the P-223 draindown pump, P-224 roughing vacuum pump, P-225A&B finishing vacuum pump and associated piping and valves required to remove borated water from the multi-assembly storage basket (MSB) prior to dry storage. The cask reflooding system includes the P-228 reflood pump, the P-229 condensate pump, and associated piping and valves required to prepare a sealed, loaded MSB for cutting of the lid and returning the MSB to the spent fuel pool (SFP).

Summary of Safety Evaluation: Affected piping of the vacuum drying system is non-QA and non-seismic. Wetted piping is A-312 TP-304 Schedule 40 stainless steel pipe which is consistent with existing design guidelines for the existing piping. Materials for both systems have been designated and ordered as augmented quality (AQ).

Service air (SA) is required to provide air to the air-powered knock-out pit condensate pump (P-226) as well as to solenoid valves 9708-S and 9710-S located on the P-225A&B pump skid. Demineralized (DI) water is required to provide a mechanical seal to drain down pump P-223 and roughing vacuum pump P-224. Seal water (at about 3 gpm) from P-223 ends up in the return line to the SFP. Solenoid valve 9722-S is provided in the DI line prior to P-223 and set to open only on P-223 activation. DI water is also required for flushing of the P-224 roughing vacuum pump prior to layup.

The MSB environment is sampled prior to unloading to detect the presence of failed fuel elements. If the analysis indicates no fuel damage, the helium environment in the MSB is vented out of the drumming station exhaust via the priming line using either the P-224 roughing vacuum or the P-225A&B finishing vacuum pumps prior to beginning the reflooding operation. If the analysis shows failed fuel elements, the MSB environment is purged by displacing it with nitrogen. Calculation 95-0103 documents acceptability of the release path via the drumming station exhaust.

System tie-ins (service air, DI water, SFP cooling, PAB drains, drumming station exhaust) meet or exceed the design requirements for those systems and are isolatable from those systems. The tie-in to the SFP cooling is from the demineralizer loop and is isolatable from the main coolant loop. The flow rate is within the 60 gpm design of the SFP demineralizer.

The VDS and CRS are outside the boundaries of ASME Section XI and are therefore subject to the requirements and administrative requirements of the Wisconsin Administrative Code, ILHR 41. Per ILHR 41.04(29) only piping which is subject to steam pressures in excess of 15 psig or water temperatures greater than 250°F is considered "power piping."

The steam pressure provision of ILHR 41.04(29) limits the "power piping" applicability to piping used to vent the steam generated during possible MSB unloading operations. The hot water temperature provision of ILHR 41.04(29) limits the "power piping" applicability to the P-223 drain down pump flow path (since the borated temperatures may exceed 250°F) and to piping used for cooling/venting during possible unloading operations.

To ensure pressure integrity, all wetted piping is fabricated per ASME B31.1-1992. Per B31.1, the welds shall be visually inspected and shall be subject to the acceptance criteria of B31.1-1992. In addition to this requirement, 10% of the drain piping shall be subject to radiographic testing with acceptance criteria defined by B31.1-1992.

DI water enters the drain down flow stream at a rate of approximately 3 gpm through the seals of the P-223 drain down pump and be delivered to the SFP. This is a very slow dilution of the SFP water. The SFP boron concentration is raised to >2850 ppm per the Certificate of Compliance for cask loading. P-223 is used after the MTC is removed from the SFP, at which time the required SFP boron concentration is >1800 ppm per TS 15.5.4. Failed shut solenoid valve CDW-9722-S provides seal water flow to P-223 only when P-223 is in use. The limited use of P-223 combined with the high initial SFP boron concentration, large pool volume, small DI water flow, the SFP high level alarm, and that the SFP remains subcritical with unborated water, eliminates SFP dilution concerns.

There is no equipment important to safety under 10 CFR 50 in the cask loading area that could be adversely impacted by the VDS and CRS. The non-seismic VSD and CRS are not positioned over required seismic equipment. (SER 95-064)

Summary of Safety Evaluation: The vacuum drying system (VDS) and cask reflood system (CRS) are not explicitly described in the ISFSI licensing basis documents. Evolutions which must be performed are described. These evolutions require the use of the VDS and/or the CRS. The VDS and CRS are designed to meet functional and design requirements.

A calculation demonstrates that the MSB meets ASME Section III Service Level A allowable stresses at a pressure of 41.6 psig (56.3 psia). Overpressurization of the MSB from SFP water is not possible. The maximum dead head pressure of the SFP pumps (P-12A&B) at the MSB is about 34 psig. Overpressurization from bottle gas is provided by a series of pressure regulators with relief valves downstream to protect the MSB should the regulators fail. If P-226 or P-229 should rupture the diaphragm, the MSB would be protected from service air pressure because the air would vent directly to the atmosphere through the other port. Overpressurization from the P-228 reflood pump is provided by relief valve CRF-9752. Overpressurization from the P-229 condensate pump and DI water is prevented by isolating dilution paths. During reflood, the flow rate is controlled and pressure is maintained at less than 41.6 psig.

The VDS and CRS are not used to mitigate accidents as discussed in the safety analysis report. The consequences of a breach of the MSB are not increased by the VDS and CRS because the source term is determined by the fuel selection criteria.

The systems are designed with adequate relief capability. To ensure pressure integrity, wetted piping is to be fabricated per ASME B31.1- 1992. Per B31.1, the welds shall be inspected visually and shall be subject to the acceptance criteria of B31.1-1992. In addition to this requirement, 10% of the drain piping shall be subject to radiographic testing with acceptance criteria defined by B31.1- 1992. Hydrostatic (or air) tests at 150% of design pressure shall also be performed. The VDS/CRS has also been successfully functionally tested during the dry runs. (SER 95-064-01)

30. MR 92-120, (Common), ISFSI. This revision includes additional information supporting the evaluation conclusions that the criticality analyses for the VSC-24 system are not adversely affected by the reduction in the MSB sleeve thickness.

Summary of Safety Evaluation: Keno criticality analyses show the 0.20" sleeve thickness has a higher Keff than the 3/16" thick sleeve in the MSB. The analyses are based on the worst-case accidental loading of all fresh 4.2 w/o enrichment fuel and worst-case geometry (e.g., all fuel moved inward to the maximum extent possible including the effect of reduced sleeve thickness). The analyses show that Keff is reduced by the thinner sleeve material. This is because replacement of steel with borated water in the array has a greater effect than the geometric effect caused by the assemblies being allowed to move closer by 0.0125". These analyses are based on conservatively low values for boron concentration (2600 ppm compared to 2850 required by TS) and moderator density (0.60 g/cc used compared to a nominal moderator density of 1.0 g/cc for water at 70°F). The lower moderator density further reduces the amount of boron in the array, which is the key factor in offsetting the geometric effect. Therefore, these analyses show that the bounding analysis for the VSC-24 criticality analysis is still based on use of 0.20" sleeve thickness. (SER 95-079-01)

31. MR 92-120 (Common), ISFSI. The change to the VSC-24 System reduces the MSB sleeve material thickness from 0.20 wall thickness to 0.1875 stock and changes the MSB support bar description to 1.45 x 28 LG, 2.0 stock.

Summary of Safety Evaluation: The changes do not affect the leak tight integrity of the MSB. The cask designer reviewed the MSB stress analysis, criticality analysis and heat transfer calculations to determine the effects of the changes. An additional calculation (CPC109.002.27) verified the change to the thickness of the MSB sleeve material does not affect the ability of the sleeve to adequately support the fuel contained in the MSB. Additionally, it has shown that the MSB internals continue to provide appropriate heat transfer and prevents criticality.

Section §11.2.3 of the safety analysis report, "MSB in Shipping Cask Drop," concludes that the MSB internals do not deform to the point that the fuel assemblies would be bound. The changes do not require a license condition to be changed because the Certificate of Compliance is not affected. (SER 95-079)

32. MR 92-120 (Common), ISFSI. The change to the VSC-24 system affects the bottom plate orientation so the bottom plate is welded inside the MSB shell rather than to the bottom of the shell as in the original design. This improves fabrication and shell roundness at the bottom and minimizes warpage of the bottom plate.

Summary of Safety Evaluation: The change in the bottom plate configuration does not change the strength, fundamental geometry of the MSB structure or leak-tight integrity of the MSB. All MSB stress analyses are based on ASME Section III requirements. The ASME Code allowable stress for a full penetration weld is the same as for base material. Thus, the ASME Code considers such welds to be as effective as base material. The weld location is not important because a full penetration weld is still being provided. The MSB stress analyses are described in the Chapter 3 of the safety analysis report. MSB stresses remain below the limits and are not affected. Additionally, the shielding and heat transfer characteristics are not affected because the weld has the same characteristics as the MSB base material. The change does not change the operation (loading, unloading, transfer or storage) of the VSC-24 system. (SER 95-080)

33. MR 92-120 (Common), ISFSI. The change to the VSC-24 system removes heat transfer angles installed below trunnions. The heat transfer angles are vertical angles located between the lead gamma shield and the outer wall of the MTC. The angles conduct heat from the lead shield, through the RX-277 neutron absorber to the outer wall of the MTC. There is no way to place RX-227 into angles below the trunnions as the angle extend up to the bottom edge of the trunnions. The keeper plate material is also changed from A36 to A514. This makes the keeper plate of similar metal to the lifting beam and helps facilitate welding.

Summary of Safety Evaluation: The original heat transfer angle design called for 32-1/4" angles equally spaced around the outer wall annulus. Removal of the angles below the lifting trunnions facilitates fabrication. Calculation WEP 109.003.07 ensures that 24 angles are adequate to transfer the heat through the RX-277 region. The calculation requires a minimum of 19-1/4" angles to produce the conductivity of 1Btu/hr-ft-F° through the RX-277 region. The removal of angles below the trunnions leaves more than the minimum 19 angles for heat transfer, thus making the calculation conservative for the actual design. The angles do not provide structural support for the MTC. The change does not affect the shielding capability of the MTC. (SER 95-081)

34. MR 92-120 (Common), ISFSI. The VCC diameter and tolerance was changed from 132.0 +/-.5 to 132.5 +.-.25. VCC wall thickness is greater than 29".

Summary of Safety Evaluation: Per calculation 95-0086, the change in the VCC diameter could theoretically increase the nominal diameter by 1/4". This could potentially add 965 lb. to the cask weight. The VCC weight considering the extra concrete mass are still below the weights used in the design of the transporter, storage pad facility, load path and cask lifting lugs. A vender evaluation indicates a maximum increase in weight by 2000 lbs. Calculations are compared to a 132" diameter cask. The slight increase in weight does not affect calculations considering VCC dead load.

Increasing the VCC concrete wall thickness affects the thermal conductivity of the cask. Due to low thermal conductivity of concrete, very little heat generated in the MSB is radiated through the concrete. Most of the heat given off by the cask is transferred through the air vents. The increase in the concrete wall thickness has a negligible affect on the thermal performance of the cask. Evaluations indicate that concrete wall temperature increases would be less than 1°F. A temperature increase of this magnitude does not affect fuel temperatures.

Radiation shielding increases as concrete thickness increases. Accident consequences are not increased by the change.

The additional VCC weight potential evaluated for the transporter, load path and VCC lifting lugs was found to be acceptable. Resulting loads are enveloped by loading used in evaluations. The probability of an equipment malfunction is not increased. (SER 95-082)

35. MR 92-120 (Common), ISFSI. To facilitate fabrication of the VSC-24 system, the MTC lifting trunnion design was changed from a hollow steel pipe filled with lead and RX-277 grout to solid steel. In addition, the welds attaching the trunnion to the MTC inner and outer shells were improved from partial penetration to full penetration welds with backing rings.

Minor changes were also made to the trunnion cover plate diameter to ensure a tight fit up to the trunnion. The weld attaching the cover plate to the trunnion is reduced in size from 3/16" to 1/8" since the cover plate base material is 1/8" thick.

Summary of Safety Evaluation: There are no accidents analyzed in the FSAR involving a heavy load drop in the primary auxiliary building. Therefore this change does not impact on the probability of an FSAR analyzed accident or the malfunction of equipment important to safety.

The change in trunnion design affects the structural strength of the MTC which is a heavy load lifting device used above the spent fuel pool (SFP). The accident analysis does not include the scenario of dropping a heavy load into the SFP or onto equipment important to safety. This condition is not considered credible since the lifting devices are designed to be single failure proof in accordance with NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants" and ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More." To be single failure proof, a factor of safety of at least 6 with respect to yielding and 10 with respect to ultimate strength must be provided. The factors of safety with the new trunnion design meet these requirements as documented by calculation CPC-109-002.09, Revision 2, which requires that the minimum yield strength of the trunnion material must be 37 ksi. Data for the material shows a yield strength of 57.9 ksi. At the possible elevated temperature of the trunnion (possibly as high as 300-350°F based on Figure 4.4-4 of the safety analysis report). The yield strength would still be well above the 37 ksi requirement. This is based on engineering judgment as the yield strength of ASTM A350 decreases from 36 ksi to 32 ksi when the material temperature increases from ambient to 300 °F.

The original trunnion shell was specified to be A36 steel and the new material is A350 Grade LF2. The new material meets the required Charpy impact requirements of 15 ft-lbs at 0°F specified in the safety analysis for the MTC. In addition, the welds between the trunnions and MTC shell walls are improved by changing the welds from partial penetration groove welds to full penetration groove welds using backing rings. The initial stress calculations for the MTC assumed partial penetration welds. The full penetration welds are known to be stronger and therefore are used as good engineering practice in this application. A revised stress calculation was not considered necessary for this change since the NUREG-0612 requirements are still met.

TS 15.4.14 specifies requirements to inspect lifting devices prior to supporting heavy loads over the SFP. The procedures for use of the MTC require this inspection to be done prior to each use of the MTC. The change has not reduced the margin of safety as defined in the basis of the TS.

Heat generated from the MSB must transfer from and through the MTC. The change in trunnion materials from a layered composite of lead, RX-277 grout and steel to solid steel improves the heat transfer through the MTC wall. This increase in heat transfer also makes the surface temperature of the trunnions hotter. Based on the temperature profile of the MTC wall shown in Figure 4.4-4 of the SAR, for the solid steel trunnions the long term equilibrium surface temperate will be between 300-350°F. This temperature was approximately 200°F for the original trunnion design. The hot surface temperature is a personnel safety concern addressed with procedural cautions.

The MTC also functions as radiation shielding when housing an MSB. The change in trunnion design changes the type of shielding provided by the trunnion from a layered steel-lead-RX-277 composite to solid steel. Calculation CPC-106.3 concludes that the dose rate at the worst case trunnion location is less than the dose rate at the MTC surface fuel midplace and is therefore acceptable. This change does not increase the consequences of a malfunction of equipment important to safety or create the possibility of a significant increase in occupational exposure. (SER 95-084)

36. MR 92-120 (Common), ISFSI. The change to the VSC-24 system eliminates the Swagelok quick disconnect from the MSB drain line. This reduces the flow restriction inherent with the Swagelok quick disconnect design. A pipe plug is substituted for the fitting to provide closure when necessary. The original design utilized quick disconnects on both the vent and drain lines. The change allows faster draining, vacuum drying and backfilling of the MSB.

Summary of Safety Evaluation: The change does not change the strength, fundamental geometry of the MSB structure, or leak tight integrity of the MSB. Additionally, the shielding and heat transfer characteristics are not affected. The change to the MSB does not change the operation (loading, unloading, transfer, or storage) of the VSC-24. (SER 95-085)

37. MR 92-120 (Common), ISFSI. The VSC-24 system MSB structural lid lifting bolt circle radius is changed from 27.0" to 26.5". This prevents possible interference between the structural lid to MSB shell weld and the lifting lug landing area.

Summary of Safety Evaluation: The accidents in the VSC-24 SAR that could possibly be affected by the change in MSB lifting lug locations are:

11.1.3, "Interference During MSB Lowering from Transfer Cask Into Concrete Cask." The change in the MSB lifting bolt circle radius of 1/2" does not have an affect on the probability of this accident. The postulated causes of this accident are improper alignment of the MTC and VCC, or foreign material present in the VCC or MTC.

11.1.5, "MSB Off-Normal Handling Load." This accident postulates that the MSB impacts the VCC during the transfer operation. The relatively small change in the MSB lifting bolt circle radius is not significant with respect to maintaining control of the MSB when it is being handled by the overhead crane. The bolt circle change has no significant effect on the calculated stresses resulting from the maximum postulated MSB to VCC impact loading.

11.2.3, "MSB In Shipping Cask Drop"/11.2.7, "Accident Pressurization." The change in location of the MSB lifting bolt holes has no impact on the stress calculations for the cask drop and accident pressurization accidents. The exact location of the bolt holes is not required as an input to these calculations.

The MSB is the confinement boundary to prevent radiological release. The changes made to the MSB lifting lug bolt circle do not significantly impact the strength or leak-tightness of the MSB. The exact location of the holes is not a required input for the stress calculations done for the MSB in shipping and accident pressurization accidents.

The accident analysis does not include the scenario of dropping the MSB during transfer of the MSB between the MTC and VCC. This condition is not considered credible since the lifting devices are designed to be single failure proof in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More." To be single failure proof, factors of safety of at least 6 with respect to yielding and 10 with respect to ultimate strength must be provided. Calculations show these factors of safety are still met with the revised MSB lifting bolt circle radius. Since the NUREG-0612 requirements are still met, the possibility of this accident is still not credible.

The change in lifting bolt location does not change the amount and location of shielding in the assembled MSB since the bolt holes are plugged with steel after removal of the lugs. The change in radiation exposure to the worker removing the lugs and installing the plugs is considered negligible due to the 1/2" change in bolting radius. (SER 95-086)

38. MR 92-120 (Common), ISFSI. The change to the VSC-24 assembly MTC cask wall assembly replaces 5" of lead with RX-277. The MTC is the transfer cask used to move the MSB between the SFP and the VCC. It is built with an inner and an outer shell of carbon steel, forming a void for shielding materials between the walls. The shielding materials consist of an inner layer of lead bricks, which provide primarily gamma shielding and an outer layer of RX-277, a grout-like material that provides primarily neutron shielding. In the original design, the lead extended to within 1" of the top ring of the MTC. The gap is left to accommodate the differential thermal expansion of the lead and steel. The RX-277 in the outer layer completely fills the available space from the bottom ring to the top ring.

Summary of Safety Evaluation: No accidents specifically analyzed in the VSC-24 safety analysis report are affected by the change to the MTC. Aspects of the MTC design that could be affected by the change include possible changes in the MTC thermal shielding and structural analyses.

The MTC thermal analysis could be affected because the RX-277 material has different heat transfer properties than the lead it replaces. The RX-277 is a worse conductor of heat, which could result in higher equilibrium temperatures for the MSB and fuel. A review of the MTC thermal analysis shows that the analysis conservatively assumed only radial heat transfer from the MSB through the MTC, and only from the active fuel region of the MSB. This change is above the active fuel region and therefore does not affect the thermal analysis.

The MTC shielding analysis could be affected because of the change in shielding materials in the MTC. Five inches of lead are removed from the top of the lead pile. The removed lead, which is primarily a gamma shield, is replaced by the RX-277 material, which provides neutron shielding. A calculation shows that the dose rates at the top ring area of the MTC remains acceptably low with the new shielding materials. The calculation did not account for the small void between the top of the RX-277 and the MTC top ring. The void is approximately 1" x 4". The presence of the void increases the expected dose rate slightly from that estimated in the calculation. The increased dose rate is not significant. Estimates on how this change affects the dose rates when compared to the original design are not possible because the dose rate in the top

ring area was not calculated in the original shielding calculation. This area was not considered because the original calculation looked at worse case locations for dose rate considerations (such as at the fuel centerline), and this area is not considered a worse case area. Because the dose is expected to remain low, this change does not have a significant effect on the shielding ability of the MTC.

A review of the MTC structural analysis determined that the analysis is not affected. The shielding materials are not part of the structure. No credit is taken for these materials, other than as weight. The weight of the MTC is slightly reduced but has no effect on the structural analysis.

The change affects the MTC, which is equipment that is important to safety. The change does not create the possibility of a malfunction of the MTC or an accident of a different type because the change does not significantly change the thermal, structural or shielding functions.

Operations involving the MTC are conducted entirely within the primary auxiliary building, which is an enclosed and controlled micro-environment. The shielding characteristics of the MTC are not significantly affected by the change. For these reasons, the change does not create the possibility of a significant unreviewed environmental impact other than those previously evaluated. (SER 95-087)

39. MR 92-120 (Common), ISFSI. The change to the VSC-24 assembly shield lid allows for its fabrication to be a single 5" plate of metal. The original design specifies two 2.5" plates welded together. The change eliminates the separate bottom support plate and improves its operation. The shield lid is installed in the MSB when it is underwater in the cask loading area. Combining the bottom support plate and the shield lid makes handling of these components much easier.

Summary of Safety Evaluation: The shield lid provides shielding from the fuel contained in the MSB. The original shielding analysis assumed a solid 5" thickness of metal in the bottom of the shield lid. That assumption is not affected by this change. The shield lid has no structural significance (except as a weight) and is not considered in structural analyses. The weight is unchanged by this design change. No accidents analyzed in the safety analysis report are affected by the change to the shield lid design.

The MSB is equipment important to safety contained in the MSB. It provides shielding for the top of the MSB. The change to the shield lid does not affect the shielding ability of the shield lid, and therefore does not change the shielding ability of the MSB.

The method of operation of placement of the shield lid is changed because the bottom support plate does not have to be separately installed. This change improves safety because of less handling of heavy components underwater.

The accepted limits of the shielding calculations are not affected by the changes. In addition, the dimensional, heat transfer, and structural characteristics of the MSB design are not affected by the change. (SER 95-088)

40. MR 92-120 (Common), ISFSI. The original VSC-24 assembly design utilized poured lead between the three inner and middle shells of the MTC. The middle shells support the poured lead. To simplify fabrication, lead bricks were substituted for the poured lead, thus eliminating the need for the middle shell.

Summary of Safety Evaluation: The middle shell is not a structural member of the MTC. Therefore, its elimination from the design does not impact the structural integrity of the MTC. While no accident or malfunctions consider the MTC, the effects on heat transfer and shielding were considered. The effects are within the bounds of the original design. The changes to the MSB do not change the operation (loading, unloading, transfer or storage) of the VSC-24. (SER 95-089)

41. MR 92-120 (Common), ISFSI. The changes to the VSC-24 assembly include a weld repair to a crack discovered on the outer radius of (one) storage sleeve corner. Also, the dimension from the top of the storage sleeves to the top of the support wall measured between 4.75" and 5.0" around the circumference of the storage sleeve assembly. The design dimension for the support wall is 5.0 +/-0.1". The vendor did not properly control the location of the support wall during fabrication.

During final assembly, the actual MSB shell outside diameter was measured at three localized areas as 62.592", 62.571" and 62.535". Other areas measured are within specification. A spider is used in the MSB top to control diameter during shipping. Installation of welding shims during MSB loading brings the shell diameter back in tolerance.

Summary of Safety Evaluation: The deviations to the MSB design during fabrication do not affect the form, fit or function of the MSB. The principle function of the MSB to serve as a high integrity, leak-tight container is not affected. The deviations from the specified tolerances have been addressed by the manufacturer to reduce the likelihood of recurrence in subsequent MSB fabrications.

The structural thermal and shielding analysis in the SAR and the design are conservative and the deviations acceptable as is. The changes do not require a license condition to be changed because the Certificate of Compliance is not affected. (SER 95-090)

42. MR 92-120 (Common), ISFSI. The change to the VSC-24 assembly is relative to the bottom plate of the MSB shell weld. The weld is changed from a single-bevel groove to a double V-groove. The revised weld type, in conjunction with dimensional changes to the bottom plate allow the bottom plate to be welded inside the MSB shell rather than to the bottom of the shell as in the original design. This improves fabrication, shell roundness at the bottom, and minimizes warpage of the bottom plate.

Summary of Safety Evaluation: The change does not change the strength, fundamental geometry of the MSB structure or leak-tight integrity of the MSB. MSB stress analyses are based on the ASME Section III requirements. The ASME Code allowable stress for a full penetration weld are the same as for base material. Therefore the ASME Code considers such welds to be as effective as base material. The weld location is not important because a full penetration weld is still being provided. The MSB stress analyses described in Chapter 3 of the VSC-24 SAR are not affected.

Additionally, the shielding and heat transfer characteristics are not affected because the weld has the same characteristics as the MSB base material. This change to the MSB does not change the operations (loading, unloading, transfer or storage) of the VSC-24. (SER 95-093)

43. MR 92-120 (Common), ISFSI Changes to the VSC-24 assembly MSB shield lid design include:

- Modifying the MSB vent arrangement to increase the vent line size through the shield lid from 1/2" to 1" and to permit the use of a ball valve in place of the Swagelok fitting. The changes improve the ability to drain and reflood the MSB during loading/unloading evolutions.

- Revising the port hole in the structural lid to accommodate dimensional changes from 4.5" to 8" and the structural lid valve cover dimensions to fit the port hole.

- Adding a seal weld between the 2.5" plates at the vent port opening to prevent water from entering between the shield lid plates.

- Recessing a threaded hole for the drain pipe in the shield lid bottom plate to make the joint stronger.

- Modifying the shield lid valve opening design to allow placement of a seal weld between the top two lid plates to facilitate placement of the seal weld.

- Enlarging the valve opening diameter in the shield lid by 0.25" to a new dimension of 4.25".

Summary of Safety Evaluation: The increase in the vent and drain line sizes has no effect on the structural analysis. The structural lid hole for shield lid access is negligible and was not modeled in the analysis. In addition, it is welded with two cover plates before placement in storage. Overall thickness of the shield lid and its steel plates was not changed. Therefore, the same amount of material is provided to resist the drop loads. Small vent openings are negligible and are not considered in the licensing calculation for structural and shield lids. The change does not affect the thermal analysis previously performed. The lid conductivity and radiation properties are not altered. No axial heat transfer through the lids is credited in the analysis.

The addition of a seal weld between the 2.5" plates at the vent port opening prevents water from entering between the shield lid plates. It is intended to seal off the RX-277 inside the shield lid. This does not impact analysis contained in the safety analysis report.

The shield lid valve opening design modification allows placement of the seal. It does not affect the licensing analysis. There is sufficient room to perform field welding between the shield and structural lids. The small hole was not included in the initial VSC-24 licensing analysis.

The shield lid provides the second confinement boundary (SAR Section 2.3-1) and is not taken credit for as a pressure boundary in Section 11.2.7, "Pressurization Accident." The structural lid forms the pressure boundary. The structural lid valve covers are double seal welded and become part of the structural lid. The ball valve on the vent line is below the valve covers so it is not considered a pressure boundary for accident analysis. RP-8 Part 3, for unloading the fuel from the cask, assumes that the valves have leaked and a hot tap method is used for initial venting. The vent openings are considered negligible for thermal and thermal-hydraulic analyses. The calculations do not consider these openings. The thermal analysis conservatively only considers radial heat transfer. No axial load heat transfer through the shield lids is credited in the analysis. The adiabatic heat-up in the MTC has not changed because the lid weight and material composition has not changed. The shielding calculations in the safety analysis report do not model the vent holes. The openings are small and the effect of size change is considered negligible. A

calculation of the cover weld analysis shows that the weld strength is very conservative. The calculation concluded that the MSB stresses, thermal analysis, structural, and shielding requirements are not significantly affected. (SER 95-094)

44. MR 92-120, (Common), ISFSI. This safety evaluation reviews the ventilated storage cask (VSC) system. The VSC system is a dry storage system utilizing a concrete storage cask, and a steel, seal-welded multi-assembly sealed basket (MSB) to store irradiated nuclear fuel. It includes: a ventilated concrete cask (VCC); 24-assembly multi-assembly sealed basket (MSB); MSB transfer cask (MTC); vacuum drying and helium (He) backfill system with a He sniffer for leak detection; and engineered cask transporter.

The MSB is sized to hold 24 Westinghouse pressurized water reactor (PWR) 14x14 fuel assemblies. The MSB is stored in the central cavity of the VCC. The cask is ventilated by internal air flow paths that allow for decay heat removal by natural circulation around the metal basket wall. The MTC is a large shielded cylinder used to hold the MSB while in the spent fuel pool (SFP) and then transports the MSB from the SFP to the decontamination area and then to the VCC inside the auxiliary building. The vacuum drying the helium backfill system is used to drain and purge the MSB once it is out of the SFP. The engineered cask transporter moves the loaded VCC from the primary auxiliary building (PAB) to the outside storage pad.

Summary of Safety Evaluation: The VSC is licensed under 10 CFR 72 Subpart K "General Licenses for storage of Spent Fuel at Power Reactor Sites." The VSC system was reviewed and approved by the NRC and has received an approved NRC Safety Evaluation Report and Certificate of Compliance. This safety evaluation complies with Subpart K Section 72.212(b)(4) that requires each facility, prior to use of the general license, to determine whether activities related to storage of spent fuel under the general license involve unreviewed facility safety questions or changes in the Technical Specifications as provided under 10 CFR 50.59.

45. MR 92-120 (Common), ISFSI. To facilitate a manufacturing change, the size and type of the cask wall-to-rail assembly weld is changed. The original design specified a continuous 5/8" fillet weld between the two structures. This fillet weld could not be achieved in a portion of the prescribed area because of MTC configuration. The rail assembly forms a tangential contact area with the cask wall assembly. This area does not allow the use of a 5/8" fillet weld due to lack of exposed base material on the rail assembly when the cask wall assembly is placed in its final position. This necessitates the use of a fillet weld of less than 5/8".

Summary of Safety Evaluation: The change in the cask wall-to-rail assembly weld design affects the structural strength of the MTC which is a heavy load lifting device used above the spent fuel pool (SFP). The accident analysis does not include the scenario of dropping a heavy load into the SFP or onto equipment important to safety. This condition is not considered credible since the lifting devices are designed to be single-failure proof in accordance with NUREG-0612, "Control Of Heavy Loads at Nuclear Power Plants" and ANSI N14.6 "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More." To be single-failure proof, factors of safety of at least 6 with respect to yielding and 10 with respect to ultimate strength must be provided. The welds between the cask wall and rail assembly are improved by changing the welds from a fillet to a full penetration groove combined with the same fillet weld. The initial stress calculations for the MTC assumed fillet welds. The full penetration welds are known to be stronger and therefore were used as good engineering practice in this application. NUREG-0612 requirements are still met and the possibility of this accident is still not credible.

TS 15.4.14, "Surveillance of Auxiliary Building Crane Lifting Devices," specifies requirements to inspect lifting devices prior to use in supporting heavy loads over the SFP. The procedures for use of the MTC require this inspection be done prior to each use of the MTC. This change has not reduced the margin of safety as defined in the basis of the Technical Specifications.

A detailed stress analysis of the cask wall-to-rail assembly weld is contained in Chapter 3 of the VSC-24 Safety Analysis Report. The maximum shear force and stress are calculated based on a 1/2" fillet weld (4 ksi), while originally specified as 5/8" fillet weld. Using the 5/8" fillet weld in the same formula results in an equivalent shear stress of 3.16 ksi. Hence the original 5/8" fillet weld was more conservative than that analyzed in the calculation. The change results in a change of the original 5/8" fillet weld to a 1/2" groove weld combined with the maximum achievable fillet weld. A conservative recalculation considers the 1/2" groove weld and 5/8" fillet weld independent of one another. In other words, in areas where the full 5/8" fillet could not be achieved it is assumed that only the groove weld existed. Using the same formula in the safety analysis report, this results in a stress of 2.8 ksi for groove weld and 3.16 ksi for the fillet weld. In areas where both of these welds existed, the stresses would be proportionally less with the maximum stress bounded by the 5/8" fillet weld (3.16 ksi).

The use of an equivalent size groove or a larger fillet weld results in a reduction in the maximum shear stress in the weld due to its larger area. The resulting shear stress seen in the weld is less than originally analyzed. This change in the weld configuration increases the strength of the weld and does not fundamentally change the geometry of the MTC structure. The weld location is unchanged. Additionally, the shielding and heat transfer characteristics are not affected by this change because the weld has the same characteristics as the MTC base material. This changes does not change the operation (loading, unloading, transfer or storage) of the VSC-24 as described in the safety analysis report and Certificate of Compliance. (SER 95-096)

46. MR 92-120 (Common), ISFSI. The safety analysis report describes a six-wheeled, towed vehicle for lifting and transporting the VSC. The Point Beach transporter is a self-propelled, tracked vehicle with a hydraulic lift beam designed to lift the VSC from the top using lifting lugs embedded in the VCC. The engine is fueled by propane contained in cylinders secured to the transporter. The power from the engine is transferred to the track drives by a hydraulic system. This same system provides power to the lift beam hydraulic cylinders.

Summary of Safety Evaluation: A dry storage transporter designed for the ISFSI is not considered important to safety since it does not lift the loaded VSC higher than 18". The VSC is analyzed for a 80" drop without breach of the MSB. Furthermore, a drop of 18" or less does not result in unacceptable damage to the VCC, MSB or fuel assemblies. The transporter is designed to move a fully loaded VSC up a 5% grade at a maximum speed in forward or reverse of 1.5 miles per hour. The braking system is fail-safe in that hydraulic pressure is required to release the drive motor brakes and propel the transporter. A separate emergency stop switch is provided to allow a person other than the operator to stop the transporter if necessary. The transporter is designed to lift the 135 ton cask no more than 18" above the ground during transport operations. A camlock device is installed on the lifting cylinders to prevent the lift beam from dropping if hydraulic pressure is inadvertently lost. During transport, the cask is restrained from swinging by a cask restraint.

The only accident or malfunction in the safety analysis considered for the transporter is 11.2.1, "Failure of all Fuel Pins with Subsequent Ground Level Breach of MSB." This hypothetical accident assumes the failure of fuel pins and breach of the MSB with a subsequent release of the gaseous fission products to the environment. The transporter is designed and fabricated to prevent impact of forces which may initiate this postulated accident. Furthermore, since this accident is bounding in terms of consequences, no other accidents or malfunctions need be considered from a safety perspective.

There are no unreviewed environmental impacts or significant increases in occupational exposure as a result of this transporter design. The transport time is longer than that assumed in the safety analysis report. However, the additional exposure received during transport is not significant when compared to the total dose required to load and move a cask. Furthermore, actual exposures are lower than estimated based upon industry experience and because of conservatism in the dose calculations. (SER 95-110)

47. MR 92-120 (Common), ISFSI. Engineering change request ECR 95-052 modifies the shape of the VSC-24 system MTC lifting yoke hooks to eliminate potential interference with the spent fuel pool (SFP) racks. The pin diameter is reduced to ease engagement with the PAB crane hook. ECR 95-240 further modifies the MTC lifting yoke hooks to allow the lifting yoke to hang straighter on the PAB crane hook. Straightening the hang minimizes potential interference with the SFP racks.

Summary of Safety Evaluation: The changes to the MTC lifting yoke do not increase the probability of occurrence of an accident or equipment malfunction outlined in the FSAR. The changes make the lifting yoke easier to operate, thereby reducing the probability of a cask handling accident or interference with the lifting yoke and other equipment.

The consequences of accidents or equipment malfunctions evaluated in the FSAR were reviewed to determine if these changes could cause an increase in the radiation exposure for these accidents. The MTC lifting yoke itself performed no shielding or containment function. Therefore, the changes do not increase the consequences of an accident evaluated in the FSAR.

An MSB cask drop in the primary auxiliary building (PAB) resulting from a lifting yoke malfunction would be an accident of a different type than previously evaluated. However, the requirements of NUREG-0612 and ANSI N14.6 are still met. Therefore, a cask drop accident evaluation if not required because the PAB crane and lifting yoke meet the requirements of NUREG-0612. No other new accident or malfunction initiators or sequences are possible because the MTC lifting yoke itself performs no shielding or containment function. (SER 95-114)

48. MR 92-120 (Common), ISFSI. The VSC-24 SAR Section 11.2.7, "Accident Pressurization" analysis of the maximum MSB internal pressure is based on 34.6 psig. This accident pressure is the result of a hypothetical breach of all fuel rods in the MSB and the acceptance criteria is based on ASME Section III, Level D criteria. Calculation shows that a maximum MSB pressure of 41.6 psig is acceptable based on the more limited ASME Section III, Level A criteria.

Summary of Safety Evaluation: The analysis shows the MSB can withstand a higher pressure. It does not affect the probability of a previously evaluated accident because the probability of initiating events have not been changed. This analysis was performed to address the possibility of higher pressure in the MSB during the reflood phase of the unloading process. The use of the Level A acceptance criteria from ASME Section III is conservative and shows that the possibility of this higher pressure does not affect the integrity of the MSB during the reflood. This analysis does not impact other equipment important to safety.

Although it is not specifically stated in the VSC-24 licensing basis, the margin of safety for confinement of the spent fuel is based on the maximum pressure that the MSB can withstand. The use of the Level A acceptance criteria from ASME Section III, is conservative and shows that the possibility of this higher pressure does not affect the integrity of the MSB during the reflood. The margin of safety for the pressure retaining capability of the MSB is preserved, based on the use of the same version of the ASME Code as used in the original design. The Certificate of Compliance states that the MSB helium backfill pressure of 14.5 psia was selected to ensure that the pressure within the MSB is within the design limits of 49.4 psia during any expected off-normal operating condition. The "expected off-normal operating conditions" being referred to did not include reflood of the MSB, because at the time of issuance of the Certificate of Compliance calculations for MSB pressure during reflood had not been performed. As MSB reflood calculations were performed, it was discovered that pressures slightly higher than the accident pressurization could occur during reflood, hence the need for the calculation. The license conditions as contained in the Certificate of Compliance remain valid. (SER 95-118)

49. MR 92-120 (Common), ISFSI. During the course of loading the first VSC-24 cask, it was discovered that the welds attaching the rails to the shell for the MTC were not load tested in accordance with ANSI N14.6. This condition was discovered when the MTC containing a loaded MSB was in the decontamination area during vacuum drying operations. The requirement to test in accordance with ANSI 14.6 is documented in the VSC-24 Certificate of Compliance requiring that the ". . . MTC is designed and fabricated as a lifting device to meet NUREG-0612."

The MTC load test conducted by the manufacturer successfully tested the trunnions, doors, door support plates and door plate to rail welds to 300% of the MTC load carrying capacity. However, because of the testing methodology used, the rail-to-shell welds were only tested to approximately 130% of the MTC load carrying capacity.

Summary of Safety Evaluation: It is possible to load the MTC rail-to-shell welds to 300% with an MSB inside. However, because the MSB is in the MTC, inspection of the interior welds is not possible during the load test and could not be performed prior to a lift of the MTC containing the MSB. Therefore, the load test of the MTC shell-to-rail welds with the MSB contained within the MTC is impractical.

It is recommended that the MSB preparations continue and that the MTC be placed on the VCC so the MSB can be safely placed in the VCC. In order for this lift to be performed in accordance with NUREG-0612, the MSB is rigged to the lifting yoke in order to provide a "dual load path." This is done by attaching the swivel hoist rings to the MSB structural lid, rigging slings over the lifting yoke and attaching it to the hoist rings. During this evolution, the MTC is carrying a loaded, dry MSB with lids.

To further ensure the safety of this operation, accessible rail-to-shell welds were non-destructively examined while the MTC is in the decontamination area. No weld defects were found as a result of this examination.

The rail-to-shell welds are essentially the same as the door guide-to-rail welds that were successfully load tested to 300%. Since both these weld groups are subjected to approximately the same stress levels and the same welding materials and processes were used, it is reasonable to conclude that the rail-to-shell welds will also pass a 300% load test.

These welds have already carried the maximum load. This occurred when the MSB with fuel and water was removed from the SFP. Since the water is removed prior to locating the MTC on the VCC, the weight of the MTC contents is reduced by approximately 10,000 pounds.

The lifting yoke is designed to support the combined weight of the MTC and MSB in accordance with NUREG-0612. Attaching the MSB slings to the lifting yoke does not affect the design, capacity or function of the lifting yoke. Therefore, the probability of a malfunction of the lifting yoke is not increased. The MSB has been analyzed to be supported or lifted from the structural lid. Since the MSB slings are installed by hand, they can not be overtensioned during installation to the point the MSB is lifted off the MTC doors. This prevents the possibility of overstressing or damaging the MSB. Therefore, the attached rigging does not increase the probability of a malfunction of the MSB. (SER 95-119)

50. MR 93-040, (Common), Service Water. MR 93-040 modifies 16 existing pipe supports to meet thermal and seismic analysis requirements. The design changes meet commitments made as a result of NRC IEB 79-14 as-build reconciliation program. The support upgrades include pipe supports on the Unit 1 containment spray (CS), component cooling water (CCW), and service water (SW) systems and Unit 2 SW system. Fifteen of the modifications are required to ensure that a seismic event does not result in pipe or support component stresses greater than Code allowable. One modification is required to prevent analyzed thermal stresses in excess of Code allowable which would be imposed on the CS piping in the event of a loss of coolant accident (LOCA) or a main steam line break.

Summary of Safety Evaluation: The current as-built piping and pipe supports for the affected piping systems are evaluated for their ability to withstand the design basis loads and stresses imposed on the systems. The evaluations include both pipe and pipe support stress analyses performed as part of the 79-14 Reconciliation program. The results show that, while unanalyzed thermal mode (due to a LOCA or main steam line break), would impose stresses on the piping or supports in excess of Code allowable values. The stresses are reduced to values below Code allowables. Code stress allowables for the affected piping and pipe supports are satisfied. Temporary support of the piping is not required to ensure system integrity. The operability of the piping systems shall remain unaffected throughout the installation process.

The design changes enhance the affected piping's capability to withstand thermal and seismic loads and are appropriate for the current conditions. This modification does not involve an unreviewed safety question. (SER 93-084)

51. MR 93-045, (Unit 1) and 93-046, (Unit 2), Condensers. The lower condenser steam erosion problem in the Unit 1 and Unit 2 main condensers was evaluated as a result of impingement steam from the steam dump valves during startup and shutdown operations. Engineering evaluation shows the need to maintain a 79" safe distance from the nozzle on the steam dump valve to the first point of contact based on measured flow, temperature and pressure. This change controls the safe distance by lowering condenser hotwell level to an acceptable range from the standpoints of erosion and safe operation. The change lowers the normal operating hotwell water level to a range of between 18-22" as referenced from 4" above the condenser floor.

Summary of Safety Evaluation: The original design level for normal operation was in the range of 22". Due to complications with the ability to maintain condensate inventory with less than optimum water make-up capabilities, an administrative decision was made to maintain the condenser hotwell water level at 25-35". Taking into consideration the improved capabilities of the water makeup system, there is less need to maintain the hotwell level at this higher range, and the concern for the degradation of the lower condenser tube bundle due to steam erosion provide additional justification of lowering it. The change in normal level range may be overridden during unusual condition (e.g., planned water treatment system outage).

The statement, "The hotwell has sufficient storage for three minutes of operation at maximum throttle flow with an equal free volume for surge protection" in FSAR 10.2-8 may be removed since actual condenser hotwell level is not an input/parameter used in computer models for Chapter 14 accident analyses. While this statement has been in the FSAR and its predecessor FFDSAR, it was apparently included in the FFDSAR following a vendor calculation. The calculation determined it would take three minutes for condenser hotwell level to decrease from its normal control band to a point where the condensate pumps would begin cavitating, assuming no condensate addition to the condenser and the condensate pumps operating at runout. This statement does not represent a regulatory requirement nor a commitment.

Lowering of the condenser hotwell levels does not effect safe operation. The action does not negate FSAR assumptions nor does it present problems different than those previously considered. (SER 94-047)

52. MR 93-049, (Unit 1), Rod Control. The rod control system post-modification tests verify that the rod control system logic cabinet timing changes are properly implemented and that the rod control system timing changes recommended to preclude movement of less than a group of control rods, have no effect on normal rod motion. The timing changes require repositioning of diodes on the printed circuit boards in the rod control system logic cabinets.

Summary of Safety Evaluation: The repositioning of the diodes does not change the functionality or the design of the boards. The timing changes have no impact on normal operation or during a transient, and will not change assumptions made in the plant safety analysis. The timing changes enhance the reliability of the rod control system and have no adverse impact on safety-related systems. The timing change is designed to preclude uncontrolled asymmetric rod withdrawal within a group.

The changes to the PC boards in the rod control system logic cabinets reduce the possibility of asymmetric rod movement during rod control system operation when certain single failures are present. However, the timing change creates some additional failure modes. The new failure modes are such that a failure of a diode on one of the decoder cards may result in rod motion failure. The failure is only in one direction and depends upon what card fails and at what step in the rod timing sequence the failure occurs. However, no new uncontrolled failure modes are created and all of the controlled failure mode rod movements continue to be within the bounds of the single rod control system malfunction presented in the current licensing basis. (SER 95-007)

53. MR 94-013, (Common), Transformers. This modification upgrades the anchorage on station service transformers 2X-13 and 2X-14. The function of 2X-13 and 2X-14 is to step down 4160 V to 480 V, feeding the B-03/B-04 safeguards buses.

Summary of Safety Evaluation: A Seismic Qualification Utility Group (SQUG) walkdown indicated the existing anchorage for 2X-13 and 2X-14 was an outlier, as defined in the program and required upgrading in accordance with USI A-46. The modification is a design enhancement to the station service transformer anchorage.

The design seismically upgrades the Unit 2 station service transformers. It includes installing two 4" diameter Schedule 80 steel pipes through the transformer bases which anchor the transformer to the concrete floor slab. Each pipe is welded to the transformer support base at two locations. The steel piping ends are welded to a plate end connection and the plate end connection is welded to a base plate which is anchor bolted with concrete expansion anchors to the concrete floor.

Phase 1 work includes anchor bolt hole drilling and is performed while the transformer is in service or energized. The anchor bolt holes are drilled in the concrete floor slab in a location near the transformer base. Phase 2 enlarges the holes in the transformer base while the transformer is in service or energized. Phase 3 includes welding of the 2X-13 and 2X-14 transformer bases. The work is done in parallel with the EDG work while 2X-13 and 2X-14 are deenergized. Precautionary measures are taken to protect the transformer tank and cooling coils from the welding arc. The electromagnetic forces (EMF) generated by the welding supply cables is negligible in comparison to the amount of EMF generated by the other cables and electrical equipment in the cable spreading room. A conservative measure is taken in order to prevent RFI sensitive equipment in the area from being affected by the welding arc. Also, due to the welding work during the third phase, the fire detection for the Halon suppression system is temporarily taken out of service to avoid a potential actuation as a result of smoke that may be generated during the welding process. Compensatory measures are taken while the Halon system is out of service. During Phase 3, some existing friction clips may need to be removed for the purpose of installing the new anchorage. This does not have a negative effect on the operation of the transformer or plant since the transformer is deenergized. The new anchorage is in place when the transformer is reenergized and in service. (SER 95-105)

54. MR 94-075 (Common), Fire Protection. This modification permanently removes the T-32A&B fuel oil storage tank automatic foam fire suppression system from service. Related alarms and signals to the Control Room remain in place. The Control Room receives signals regarding the presence of a fire at the fuel oil storage tanks, but no actuation methods of fire suppression are available from the Control Room. Fire suppression at the fuel oil storage tanks is accomplished by portable foam generating equipment manned by the fire brigade.

Summary of Safety Evaluation: The original safe shutdown analysis took credit for fuel from the fuel oil storage tanks to fuel the EDG diesel-driven fire pump and G-05 combustion turbine in the event of a fire and loss of offsite power. Adequate fuel supplies are available in the day tanks for the plant to reach hot shutdown. However, the fuel in one of the two fuel oil tanks would be needed to reach cold shutdown, with the onsite EDGs providing power, within 72 hours as required by Appendix R.

The fuel oil supply arrangement for the EDGs was modified as part of the new EDG building addition. Following U2R21 the EDGs are supplied by the fuel oil storage tanks and transfer pumps located in the new EDG building. There are two redundant fuel oil storage tanks buried below the new EDG building that each have the capacity to supply fuel to any of the four EDGs for an Appendix R fire. The outdoor fuel tanks (T-32A&B) and transfer pumps (P-70A&B) remain the primary fuel supply for the diesel fire pump (P-35B) and combustion turbine (G-05), but no longer provide a safety-related function. This system is maintained as a third backup to the new EDG fuel oil tanks installed to supply all four EDGs.

For a fire at the outdoor tanks, plant emergency power can be provided from any of the four EDGs with available fuel supplies from the new tanks buried below the EDG building. Adequate fuel for P-35B is maintained in the day tank located in the pumphouse. Therefore, a fire in the storage tanks has no effect upon plant operation. The fire fighting objective is to contain and control exposures to other plant property and equipment using the plant fire brigade and local fire department.

FSAR 9.6.1 states "A fixed foam system is provided to the two above ground fuel oil storage tanks to combat possible Class B fires." NRC Safety Evaluation Report 4.3.1 (6) states, "A fixed three percent concentrate protein foam extinguishing system is provided to the two above ground fuel oil storage tanks. Operation of the foam systems provided by actuation of rate-of-rise pneumatic detectors located under the tank cover. The foam concentration tank, proportioner, control valves and a foam hose reel are located at the fuel oil pumphouse. We conclude that the foam system is adequate for its intended use and satisfies the objectives identified in Section 2.2 of this report and is, therefore, acceptable." NRC Safety Evaluation Report 5.15.4 states, "The two fuel oil storage tanks in the yard area are protected by an automatic fixed foam extinguishing system." Branch Technical Position 9.5-1 requires that diesel oil fuel tanks of >1100 gallon capacity be located at least 50′ from any building containing safety-related equipment.

Protection of the fuel oil storage tanks, outlined in Appendix A to BTP 9.5-1, was reviewed and approved in the Fire Protection Evaluation Report (FPER). However, the fixed foam suppression system, although installed and referenced at the time of the FPER was not necessary to meet the guidelines of Appendix A to BTP 9.5-1. The location and arrangement of the tanks more than 260′ from the main plant structure and greater than 60′ from the fuel oil pumphouse containing safety-related fuel oil transfer pumps is adequate to satisfy the Appendix A guidelines without the fixed foam system. Therefore, removal of the fixed foam system from service does not change the current fire protection license basis, although it does change the current FPER and FSAR descriptions.

The tanks still have detection from rate compensated detectors that alarm locally and in the Control Room. Fire fighting water supplies are available from fire hydrants 21, 22 or 23 located greater than 50′ from the burn area. Portable foam generating equipment is located with brigade fire fighting equipment. Hose lines could be run from the hydrants to the tank area from any of these hydrants. The portable suppression system is the primary fire suppression system with the backup suppression system being the Two Creeks Fire Department. Local offsite fire brigades were made aware of the changes. (SER 95-078)

55. MR 94-019, (Unit 1), 4160 V Electrical. The modification modifies the 4160V bus 1A-05 degraded grid voltage (DGV) time delay scheme to differentiate between DGV relay operation and DGV relay operation coincident with an SI signal. This allows for a short time delay during an SI, which protects safety-related equipment from damage or tripping and a longer time delay during normal operation which allows for voltage transients during the starting of large motors such as the reactor coolant pumps.

Summary of Safety Evaluation: Auxiliary instantaneous relays parallel to the existing time delay relays are installed in the DGV scheme. The energizing of these auxiliary relays provides a similar 2/3 logic to the tripping coil but requires an SI signal from either unit to trip.

Tie-ins for the new relays, test switches and tripping logic to the existing circuits is performed while bus 1A-05 is deenergized. During this period, breaker 1A52-57 is racked out and undervoltage protection for bus 1A-05 is not required. To facilitate connections to the existing circuits, fuses are pulled which isolate these circuits while work is performed. Tests are performed by cycling breaker 1A52-57 in the test position. This ensures that new circuitry functions as intended and that the existing circuitry has not been compromised.

Degraded bus voltage is not an initiating event for accidents previously analyzed in the FSAR. The new 2/3 degraded voltage tripping logic concurrent with an SI signal is installed to bypass the existing time delay for tripping during an SI. Therefore, the new tripping logic and time delay setpoints do not increase the probability of occurrence of an accident previously evaluated in the FSAR.

The modification ensures proper voltage remains available to the equipment important to safety on these buses. Both the long and short time delay setpoints prevent these buses from operating in a sustained degraded voltage condition which could potentially damage or isolate equipment important to safety. In addition, final tie-ins to 1A-05 circuits is performed while the bus is out of service. This modification does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR.

The degraded grid voltage protection scheme is not discussed in the Basis for TS 15.3.5. However, it is required to ensure engineered safeguards equipment remains available during degraded voltage conditions. The changes do not decrease the availability of this equipment to perform its function. Therefore, the margin of safety is not reduced. (SER 95-015)

56. MR 94-020, (Unit 2), 4160 V Electrical. Quality Condition Report QCR 94-003 identified a potential for the degraded voltage relay protection scheme time delays to be inadequate. It was noted that the protection loop will not act to isolate the safeguards buses for a period of ≈50 seconds following relay dropout. If an accident occurred while voltage was in the band between the first and second level protection setpoints, it is conceivable that most of the safeguards equipment load sequencing could stall or damage safeguards loads. Since degraded grid voltage would affect both trains, a common mode failure concern exists.

To address this concern, the time delay associated with the degraded voltage scheme is reduced to ≈10 seconds. To avoid tripping the relays during a voltage transient caused by the starting of large motors, starting the reactor coolant pumps (RCPs) requires the time delay on the degraded voltage relays to be temporarily increased.

The modification changes the 2A-05 and 2A-06 4160 V bus time delay schemes to differentiate between diesel generator voltage (DGV) relay operation and DGV relay operation coincident with a safety injection (SI) signal. This allows for a short time delay during an SI, which protects safety-related equipment from damage or tripping and a long time delay during normal operation. This allows for voltage transients during the starting of large motors such as the RCPs. This is accomplished by installing auxiliary relays parallel to the existing time delay relays in the DGV scheme. The energization of these auxiliary relays provides a similar 2/3 logic to the tripping coil, but requires an SI signal from either unit to trip.

Summary of Safety Evaluation: Tie-ins for the new relays, test switches and new SI concurrent with 2/3 degraded voltage tripping logic have the following interim conditions:

Relays and test switches in the dc circuitry: The new Westinghouse Type BFD relays are installed in parallel with the existing time delay relays. In addition, the new test switches are installed in parallel with the degraded voltage relay contacts above the time delay relays. To facilitate connections to the existing circuits, fuses are pulled to isolate one of the three trip channels. During this condition, jumpers are installed to place the channel into trip, therefore temporarily converting the 2/3 logic to a 1/2 logic. Independent verification assures continuity of the jumpers and removal of the jumpers after placing the channel back in service. In addition, in-process testing is performed to assure the channel once it is placed back in service, properly performs its design function before taking the next channel out of service.

SI concurrent with 2/3 degraded voltage tripping logic: Temporary jumpers are used to maintain the existing circuitry while tie-ins for the new circuit are performed. Independent verification assures continuity and removal of the jumpers. In addition, testing is performed to assure that the new and existing tripping logic perform its intended design functions.

Prior to the modification Train B safeguards equipment required for Unit 2 operation is verified functional. No Unit 2 or common Train B testing that places Unit 2 or common Train B safeguards equipment at risk is performed. Service water operability shall be maintained in accordance with TS 15.3.3.D. No more than one service water pump in the opposite train (Train B) shall be out of service during this work. Also, G-01 EDG shall be operable. If the primary system is at a reduced inventory condition, the work will not be performed. If Unit 2 is in a condition requiring operation of an RHR pump, the Train B RHR pump should be used during this work. If relay calibration on Train B safeguards systems is in progress, the modification is not performed. During the work bus 2A-05 shall be supplied from its normal offsite power supply via 2X-04 transformer.

The new 2/3 degraded voltage tripping logic concurrent with an SI signal is installed to bypass the existing time delay for tripping during an SI. Therefore, the new tripping logic does not increase the probability of occurrence of an accident previously evaluated in the FSAR.

Adequate measures are taken to ensure proper voltage remains available to the equipment important to safety on the buses. The modification and time delay setpoints prevent the buses from operating in a sustained degraded voltage condition which could potentially damage or isolate equipment important to safety. The new 2/3 tripping logic concurrent with an SI does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR.

In addition, the design helps ensure proper voltage remains available to the equipment important to safety on the buses. This prevents the buses from operating in a sustained degraded voltage condition during an SI and potentially damaging equipment important to safety. During installation, the tripping logic is converted to a more conservative 1/2 logic to ensure that protection is maintained while one channel is taken out of service.

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. The changes do no decrease the availability of this equipment to perform its function; therefore, the margin of safety is not reduced. (SER 94-045)

57. MR 94-041*B, (Unit 1), Safety Injection. The modification cuts and caps drain valve 1SI-D-33 and the drain line from the Unit 1 safety injection (SI) and containment spray (CS) test line piping on primary auxiliary building (PAB) El 8'. This precludes fatigue damage repair.

Summary of Safety Evaluation: The drain line is not isolable from the Unit 1 refueling water storage tank (RWST) and requires a freeze seal if repair of the drain line is necessary. The drain line is similar to the Unit 2 drain line on the SI/CS test line piping which experienced cracking and leakage as a result of fatigue damage. The 3/4" drain line and valve is cut and capped close to the 6" run pipe to preclude a fatigue damage repair. The RWST is drained to allow work without a freeze seal. If a future need arises to drain the SI/CS test line, the cap on the drain line can be removed. This activity is similar to MR 94-041*A.

The work does not affect the operation of the safety injection (SI) pumps, containment spray (CS) pumps, or SI/CS test line. The potential for flooding is not increased because the RWST is drained and isolated. The post-installation verification includes a dye penetrant examination of the drain line cap fillet weld.

Removal of the drain valve from the drain line does not adversely impact the piping stress analysis for the Unit 1 SI/CS test line because removal reduces stresses due to the removal of cantilevered weight. The modification is on safety-related piping and requires QA material. The welders and weld procedure for the fillet weld are certified in accordance with ASME Boiler and Pressure Vessel Code Section XI. (SER 95-011)

58. MR 94-066, (Unit 2), Safety Injection. This modification performs two upgrades to the nitrogen supply piping to the safety injection (SI) accumulators. The first portion installs a testable containment isolation valve (CIV) 2SI-834D associated with Penetration 14c. A check valve and test connection/drain valve, 2SI-01427 is installed inside containment to allow for Type C local leak rate testing (LLRT) in accordance with 10 CFR 50 Appendix J criteria. The second portion adjusts the existing piping and 2SI-846 to provide clearance and prevent the valve body from rubbing on nearby component cooling (CC) water piping.

Summary of Safety Evaluation: The new valves, fittings and piping meet original specification requirements. The materials are compatible with existing system materials and meets the required Code and rating requirements as well as containment isolation valves and seismic requirements.

The consequences of an accident are not increased by the installation. Installation of the new check valve and test connection does not affect the integrity of the nitrogen supply to the SI accumulators. It does not adversely affect Penetration 14c nor does it adversely affect the valves, piping or components associated with the nitrogen supply line to the SI accumulators.

During installation, the nitrogen supply line to the SI accumulators is isolated to prevent an analyzed accident from occurring. The unit is in cold shutdown during installation. Containment closure checklist requirements are followed during installation.

Foreign material exclusion (FME) is accounted for during installation. Controls prevent materials from entering the system while it is open. The installation does not affect the normal operation of the nitrogen supply to the SI accumulator system. Acceptance criteria and testing is via leakage testing of the containment isolation valve. Non-destructive examinations are performed to ensure system integrity. (SER 95-098)

59. MR 94-083 (Unit 1) and MR 94-084 (Unit 2), Primary Sampling. The modifications retube the 3/8" stainless steel delay coil associated with SC-955 to place it upstream of SC-955. An evaluation identified that the SC-955 valve is missile protected; however the downstream delay coil is vulnerable to surge line failure. Failure of the downstream delay coil renders SC-955 ineffective as a containment isolation valve (CIV). Retubing the delay coil to place it upstream of SC-955 allows SC-955 valve to qualify as a CIV.

Summary of Safety Evaluation: The FSAR classifies SC-955 as a root valve for the reactor coolant hot leg sampling system. According to the FSAR, the original intent of SC-955 was to function as a root valve rather than a containment isolation valve (CIV) for Penetration 28A. This modification places the existing delay coil upstream of SC-955 and increases the amount of tubing ahead of SC-955. This reduces the effectiveness of SC-955 when viewed as a root valve. This is because there is additional tubing upstream of SC-955 and ultimately outside the possible isolation capabilities of the valve.

Two additional manually operated valves, SC-954 and SC-954A, are upstream of SC-955 and its delay coil. If the RCS sampling line upstream of the SC-955 valve requires isolation, SC-954A can be manually isolated, as it is located on the El 10' platform near SC-955.

The only credible failure of the delay coil that requires SC-955 to be used as a root valve also requires it to be used as a CIV. There would be damage to the delay coil caused by failure of the surge line. The problems present in the event of a surge line failure would be of a greater magnitude than those caused by a coincident break of the delay coil. Therefore, no concern exists should the delay coil be damaged a surge line break.

Approximately 50' of 3/8" tubing currently exists upstream of the SC-955. By adding the length of tubing which comprises the delay coil, it increases the area of tubing that is not isolable by SC-955. However, considering the length of tubing already exposed, the addition of the delay coil is not large enough to cause additional problems.

The modification is performed during a refueling outage. The RCS hot leg line in the primary sampling system is isolated while the work is performed. (SERs 95-018, 95-099)

60. MR 94-086 (Unit 1), 4160 V Electrical. The modification provides the design for the old 1A-06 switchgear (Cubicles 62-66) to become a part of the 1A-05 switchgear (Cubicles 57-61).

Summary of Safety Evaluation: The modification includes installation of a new loss of voltage relay, 1-273/A05 (ITE-27D), between Phases A and C in the 1A-05 PT circuit to allow the design of a 2/3 undervoltage scheme. In addition, a new test switch is added to bypass this new relays contacts in the new 2/3 undervoltage scheme. Test points are also revised in the new undervoltage scheme as required. It also changes the existing loss of voltage follower relays to the Westinghouse Type NBFD65NR and mount the relays inside the 1A52-61 cubicle.

The existing lockout relay on 1A-05 to an electroswitch Type 7807D is changed. This new relay provides a sufficient number of contacts to allow both 1A-05 and old 1A-06 lockout circuits to be wired to it. In addition, a new test switch is installed to isolate the lockout relay and its associated contacts.

The modification removes the 125 Vdc breaker 1A52-61 control components (x and y coils, auxiliary contacts, secondary receptacle and fuses). This breaker becomes a "dummy" breaker. It installs new shorting terminal blocks in each 4160 V cubicle 57-61 and 63-66 to allow shorting of the current transformers if desired for maintenance, testing and safety.

New fuses are installed in 4160 V cubicle 1A52-61 to isolate the 1A-05 synchronizing and voltmeter circuit from 4160 V bus 1A-05 degraded voltage and undervoltage relays supplied by the potential transformer. A new fuse monitor relay is also installed in parallel with the new 2/3 undervoltage scheme to allow indication of loss of dc control voltage to this scheme. A new alarm relay is installed in series with the 4160 V bus 1A-05 new lockout relay and in parallel with the differential relays to provide annunciation upon pick-up or failure of the lockout relay (1-86/A05).

The 1A-05 breaker control circuits, alarm circuits and diesel generator starting circuits are modified to control installation of new 4160 V 1A-05 undervoltage follower relay contacts, removal of the old 1A-06 undervoltage relays and control 1A-05 and old 1A-06 becoming one bus (1A-05).

The design changes require that the exiting switchgear door (1A52-61) be removed and a new door (taken from cubicle 1A00-38) be mounted with the new relays, test switches and test points pre-fabricated on the new door. Installation is performed on deenergized buses (1A-0" and old 1A-06) during U1R22. Testing requirements are identified for the installation and are a condition of acceptance for the modification.

A 4160 V bus undervoltage is not an initiating event for accidents previously analyzed in the FSAR. This modification implements changes to portions of the safeguards and dc electrical system for 4160 V buses 1A-05 and old 1A-06. The 1A-05 bus protective relaying does not impact 1A-01 bus or 1A-02 bus loads (e.g., main feed pumps, reactor coolant pumps). The changes do not introduce new possibilities for failure of equipment that can initiate accidents. The changes do not increase the probability of the initiating events listed in the FSAR.

Components installed are seismically verified using the SQUG methodology. Components are comparable to existing components used in the safeguards switchgear.

The undervoltage relays are not discussed in the Basis for TS 15.3.5. However, they are required to ensure engineered safeguards equipment remains available during undervoltage conditions. These changes do not decrease the availability of this equipment to perform its function. (SER 95-044)

61. MR. 94-087, (Unit 2), 4160 V Electrical. The modification provides the design for the old 2A-06 switchgear (Cubicles 67-71) to become a part of the 2A-05 switchgear (Cubicles 72-76).

Summary of Safety Evaluation: The design changes require that the existing switchgear door (2A52-72) be removed and a new door be mounted with the new relays, test switches and test points prefabricated on the new door.

Installation is performed on deenergized buses (2A-05 and old 2A-06) during U2R21. Test requirements are a contingency of acceptance for the modification.

The modification implements changes to portions of the safeguards ac electrical distribution system. The changes do not introduce any new possibilities for failure of equipment that can initiate accidents. Therefore, these changes do not increase the probability of any of the initiating events.

The original design of the undervoltage protection scheme was based on two channels per bus, with the trip of one channel causing the protection actions. This design changed to 4160 V bus 2A-05 including using three channels per bus with the trip of any two channels causing expected protection actions. The change is an improvement over the original design because an inadvertent trip of a single channel does not cause the protection actions. When a single channel is taken out of service for testing, maintenance, or calibration it can be placed in the trip condition (via its respective test switch) to allow actuation of the protection function by the trip of either of the remaining operable channels (1/2 logic).

Materials used, with the exception of the alarm relays and its mounting bases, are QA. The alarm relays are not QA because they do not serve a safety-related function nor does failure affect the function of any safety-related equipment. The alarm relay mounting bases are not QA because the relays they serve are not QA and the mounting bases are inherently rugged. Components installed are seismically verified using the SQUG methodology. Components used are comparable to existing components used in the safeguards switchgear.

The new 2A-05 design and undervoltage scheme functions to prevent important to safety equipment on these buses from operating in an undervoltage condition. Thus, this equipment is available to mitigate the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR and the consequences are not increased.

The FSAR does not indicate that the existing undervoltage relays and associated equipment cause or affect the probability of an accident evaluated in the FSAR. The addition of this equipment and the new undervoltage 2/3 logic does not change this. The undervoltage relays are not discussed in the Basis of TS 15.3.5. However, they are required to ensure engineered safeguards equipment remains available during undervoltage conditions. These changes do not decrease the availability of this equipment to perform its function. (SER 95-077)

62. MR 94-096 (Unit 1), Reactor Coolant. The modification removes existing hot leg and cold leg loop motor-operated drain valves RC-503, RC-541, RC-598 and RC-599.

Summary of Safety Evaluation: RC-503, RC-541, RC-598, and RC-599 are rarely used. If the need to drain the legs in the area of these valves occurs, alternate paths to accomplish the draindown can be used. In the past, problems with shutting the valves occurred which caused the unit to go off-line to make valve adjustments. RC-598 and RC-599 are no longer included in the MOV program because they are not used.

The valves are cut free from the lines. RC-503 and RC-541 are disposed of, while RC-598 and RC-599 are salvaged for parts. When the valves are removed, the lines are capped on the reactor coolant (upstream) side of the valve. On the downstream side, the system changes to waste disposal. A blind flange is installed at this location for possible future use. Leakoff lines from these valves to the reactor coolant drain tank (RCDT) are also capped.

The wiring and cables associated with the MOVs are removed from the valves. They are also lifted and taped at the terminal block and the motor control center. The control switches and lights are removed from the control board in the Control Room.

During this work, the reactor vessel is to be drained to 3/4 pipe and the reactor coolant system is vented to the atmosphere. The reactor coolant drain tank (RCDT) has been vented, purged and isolated. The breakers for the two MOVs (1B-32-5C and 1B-42-5C) are tagged open. The unit is defueled during the work. (SER 95-016)

63. MR 94-098 (Unit 1), Reactor Coolant. The modification removes damaged valve RC-558, reactor coolant Loop B cold leg to CVCS letdown. The disc had broken free from the stem and no longer properly operated as an isolation valve.

Summary of Safety Evaluation: RC-588 isolates RC-427 MOV from the RCS if work needs to be performed on RC-427. RC-558 is a normally open valve when the unit is at power. RC-558 offers a single isolation boundary for work on RC-427. If work would be done at power, the procedure states that double isolation is required. Since there is only a single isolation valve available, work is not to be performed on RC-427 at power. Therefore, if work is required on RC-427, it must be performed while the unit is shutdown and drained to 3/4 pipe.

The modification is performed while in cold or refueling shutdown with the RCS drained below the elevation of RC-558 (3/4 pipe). The RCS is also vented to the atmosphere. Letdown line MOV RC-427 is also isolated during the installation. The modification is scheduled for installation while the unit is defueled.

LI-447B is out of service during the installation of this modification. Alternate level indication is available to perform this function.

Removal of the valve is routine from a mechanical work standpoint. Radiation exposure minimization and awareness is accomplished through an ALARA review, pre-job brief and pre-job planning. (SER 95-017)

64. MR 95-008 (Unit 2), Reactor Protection. The modification installs additional safety-related anchorage for reactor protection cabinets 2C-151 through 2C-155, 2C-161 through 2C-165, and safeguards cabinets 2C-156 through 2C-157 to the support floor slab. The reactor protection cubicles are integrally bolted together to form one cabinet of five cubicles for each set of reactor protection cubicles listed. Safeguards cubicles are integrally bolted together to form one cabinet of two cubicles for each set of safeguards cubicles listed per unit.

Summary of Safety Evaluation: The modifications enhance the structural integrity of reactor protection cubicles 2C-151 through 2C-155, 2C-161 through 2C-165 and safeguards cubicles 2C-156 through 2C-157 by installation anchorage at the base of the affected cabinets. Reactor protection cubicle 2C-151 through 2C-155, 2C-165 and safeguards cubicles 2C-156 through 2C-157 are located in the control building cable spreading room on El 26' between Columns D-F and 10-13.

The additional anchorage of the reactor protection cabinets and safeguards cabinets provided by the modifications are to address concerns identified by the seismic review team walkdown. The walkdowns identify and help resolve issues identified in NRC Generic Letter 87-02 and USI A-46.

Reactor protection cubicles 2C-151 through 2C-155, 2C-165 and safeguards cubicles 2C-156 through 2C-157 are not taken out of service or deenergized except as allowed by refueling shutdown procedures and conditions to perform the work outlined in this modification.

TS 15.3.5 allows the engineered safeguards to be inoperable during refueling shutdowns. Procedure ICP 10.001, performed prior to each outage, bypasses and blocks safeguards and AMSAC systems to prevent accidental remote actuation of these systems. Also, during cold shutdown conditions, reactor protection logic equipment located in reactor protection cubicles, is not in service. ICP 10.001 will be in effect and the unit will be in cold shutdown prior to the start of this modification and during its installation. Performing the modification when these conditions are achieved and implementing the work control practices in the installation work plan minimizes the possibility and the impact of accidental bumping of relays and other sensitive equipment within the affected cubicles during the installation. Precautions are taken during installation to prevent metal filings or debris from falling into or on the equipment located within the affected cubicles. Therefore, operability of reactor protection cubicles 2C-151 through 2C-155 and 2C-161 through 2C-165 and safeguards cubicles 2C-156 and 2C-157 and equipment important to safety in the location of the modification are unaffected. (SER 95-076)

65. MR 95-014 (Unit 1), Safety Injection. The modification relocates existing test points in the safeguards circuits in panels 1C-157 and 1C-167. These test points verify continuity through various selector switches in the safeguards circuits. In addition, two additional test points are installed in SI panels 1C-157 and 1C-167 to facilitate voltage measurements of the 125 Vdc system.

Summary of Safety Evaluation: The relocated test points do not affect the function of the safeguards circuits. They verify continuity through various selector switches in the circuits. The new test points do not affect the function of the safeguards circuits and are used to facilitate voltage measurements of the 125 Vdc system. Installation of the new test points and mounting plate is seismically qualified in accordance with SQUG G.I.P. Post-modification testing verifies that the modification and existing circuits function as designed.

Installation is performed while Unit 1 is defueled and requires the deenergizing of the 125 Vdc control power to one or both trains of Unit 1 safeguards protection. During this condition, automatic functions for Unit 1 are not available, but safeguards equipment is able to be manually operated from the Control Room if needed. During this condition, Unit 1 safeguards system operability is not required per TS. Adequate precautions are taken to assure that equipment in SI panels 1C-157 and 1C-167 is not bumped resulting in an accidental relay actuation; however, since the safeguards system is not required to perform any function during this time the consequences of an accidental relay actuation are not a safety concern. Unit 2 safeguards equipment is not affected.

Service water is a shared system between both units. While one or both trains of the Unit 1 safeguards control power is inoperable, Unit 1 service water automatic actuation from that safeguards train is inoperable, since an automatic start of service water pumps, or service water isolation does not occur from the deenergized safeguards train(s). Safeguards actuation of service water for Unit 1 is not required to be operable by TS while Unit 1 is defueled, however service water on an emergency diesel start is required for Unit 2 operation. A TS interpretation exists for use, if required, for Unit 2. No automatic functions of Unit 2 are affected.

During U1R22, the plant is in unusual alignments to facilitate work associated with the emergency diesel generator (EDG) project. While 4160 V bus 1A-06 is out of service, service water pumps P-32A, D, E, and F are operable with normal and emergency power sources. P-32B and C are available (but not operable) for alternate shutdown. While bus 1A-05 is out of service, P-32A and F are operable with normal power because of G01 EDG LCO. P-32D and E are considered operable with normal power and

emergency power from G-04. While the Unit 1 safeguards control power is deenergized, the normal service water pump start on a Unit 1 undervoltage signal is disabled (when Train B safeguards power is out of service, automatic start of P-32C is disabled). Therefore, Unit 2 powered service water pumps P-32D-F are running to preclude effects on normal service water operation because of loss of offsite power to Unit 1. The SW pumps start on a Unit 2 undervoltage signal or a Unit 2 SI signal, therefore SW system operability for Unit 2 is not impaired.

While the Unit 1 Train A safeguards control power is inoperable, Train A auxiliary feed actuation for Unit 1 is inoperable, since no automatic start of the auxiliary feedwater pumps or automatic valve alignment occurs from the deenergized train(s). Safeguards actuation of auxiliary feedwater for Unit 1 is not required by TS while Unit 1 is defueled. Start of P-38A on a Unit 2 safeguards signal is unaffected. No automatic functions of the operating unit are affected. (SER 95-051)

Summary of Safety Evaluation: If both trains of Unit 1 safeguards control power is deenergized, Unit 2 powered P-32D-F service water pumps are running to preclude any effects on normal service water operation due to a loss of offsite power to Unit 1. If only one train is deenergized, a TS interpretation is provided for use. The SW pumps start on a Unit 2 undervoltage signal or a Unit 2 SI signal, therefore SW system operability for Unit 2 is not impaired. (SER 95-051-01)

66. MR 95-017 (Unit 1) and MR 95-018 (Unit 2), Safety Injection. The modifications replace the existing soft, resilient-type seating surfaces in the SI-854A&B check valves. The work involves the removal of the existing check valve disc was installed via MRs 90-260 and 90-261, and replaces the discs with the original discs which had been removed from the valves. The modifications return the Crane Aloyco check valves to their original design condition.

Summary of Safety Evaluation: The modifications resolve dimensional manufacturing errors in the SI-854A valve which allowed the swing arm and disc to be too low in the valve and ultimately not allow the valve discs to contact the seating surface. A shim is installed between each valve body and bonnet to raise the arm and disc up to properly meet the seating surface.

The disc and seat of both SI-854A&B are machined true and flat before installation into the existing valve. This aids properly seating the valves and reducing leakage to within acceptable limits.

In addition, the modifications also include a 3/4" vent and drain valve between SI-854A&B and SI-856A&B. This line is designed and installed within the bounds of engineering change request ECR NE-91-323. ECR NE-91-323 addresses a standard design for the installation of vent and drain valves in the SI/CS/RHR systems. Bechtel Calculation 8-5 addresses the piping displacement due to dynamic affects including seismic. Although a considerable amount of interference exists in the area, the rigidity of the piping structure precludes possible impacts between the vent and drain piping and the surrounding structures. (SER 95-052)

Summary of Safety Evaluation: In addition to providing the safety analysis for MR 95-017, this revision serves as a clarification and update. MR 90-260 installed soft, resilient type seating surfaces in the 1SI-854A&B check valves. SER 94-012 references the design temperature and pressure of the system as 600°F and 600 psig. The statement however was not clear. The pressure and temperature rating of this system is restricted by its lowest rated portion. The valve has a pressure/temperature rating of 400°F at 515 psig per its 300 lb rating classification. Upon inspection of the valves after one year of service, they were found to have degraded rubber in the seating surfaces. MR 95-017 removes the existing soft seating surfaces from the check valves and replaces them with the original hard-faced check valve discs. (SER 95-052-01)

Summary of Safety Evaluation:. In addition to providing the safety analysis for MR 95-018, this revision serves as a clarification and update. MR 90-261 installed soft, resilient type seating surfaces in the 2SI-854A&B check valves. SER 94-012-01 references the design temperature and pressure of the system as 600°F and 600 psig. The statement however was not clear. The pressure and temperature rating of this system is restricted by its lowest rated portion. The valve has a pressure/temperature rating of 400°F at 515 psig per its 300 lb rating. Upon inspection of the valves after one year of service in Unit 1, they were found to have degraded rubber in the seating surfaces. JCO 95-02-01 allowed continued operation of Unit 2 until the valves could be replaced, by U2R21 at the latest. This modification removes the existing soft seat surfaces from the check valves and replaces them with the original hard-faced check valve discs. (SER 95-052-02)

Summary of Safety Evaluation: This evaluation supersedes SER 95-052-02 and describes actual testing performed as acceptance of MR 95-018. IT-535 had originally been documented as the acceptance test for this modification. However, an operations work plan was used in place of IT-535 for acceptance testing. On May 1, 1995, the work plan was reviewed and a safety evaluation screening performed. It determined that using the work plan in place of IT-535 was not a safety significant issue and was therefore acceptable. (SER 95-052-03)

67. MR 95-032 (Unit 2), Safety Injection. The modification relocates test points in the safeguards circuits in panels 2C-157 and 2C-167. The test points are used to verify continuity through various selector switches in the safeguards circuits. Two additional test points are installed in safety injection (SI) panels 2C-157 and 2C-167 to facilitate voltage measurements of the 125 Vdc system.

Summary of Safety Evaluation: The relocated test points do not affect the function of the safeguards circuits and are used to verify continuity through various selector switches in the circuits. The addition of a two-pole banana jack does not affect the function of the safeguards circuits and is used to facilitate voltage measurements of the positive-to-negative 125 Vdc control power. The installation is seismically qualified in accordance with SQUG GIP. Post-modification testing verifies that the modification and existing circuits function as designed.

Installation occurs while Unit 2 is defueled (U2R21) and requires the deenergizing of the 125 Vdc control power to one or both trains of Unit 2 safeguards protection. During this condition, automatic functions for Unit 2 are not available, but safeguards equipment is able to be manually operated from the Control Room. During this condition, no Unit 2 safeguards system operability is required per TS. Adequate precautions are taken to assure that equipment in safeguards panels 2C-157 and 2C-167 is not bumped, resulting in an accidental relay actuation. However, since the safeguards system is not required to perform a function during this time, the consequences of an accidental relay actuation are not a safety concern. Unit 1 safeguards equipment is not affected.

Service water is a shared system between both units. While one or both trains of the Unit 2 safeguards control power is inoperable, Unit 2 service water automatic actuation from that safeguards train is inoperable, since no automatic start of service water pumps will occur from the deenergized safeguards train. Safeguards automatic actuation of service water for Unit 2 is not required to be operable per TS while Unit 2 is defueled. Unit 1 safeguards functions including service water are not affected by this modification and will operate as designed.

While the Unit 2 Train A safeguards control power is deenergized, a Unit 2 Train A auto start of service water pumps P-32A, P-32B, and P-32F is disabled. While the Unit 2 Train B safeguards control power is deenergized, a Unit 2 Train B auto start of service water pumps P-32C, P-32D, and P-32E is disabled. The service water pumps will start on a Unit 1 undervoltage signal or a Unit 1 safety injection signal, therefore service water system operability for Unit 1 is not impaired.

While the Unit 2 Train A or B safeguards control power is inoperable, the Train A or B auxiliary feedwater actuation for Unit 2 is inoperable, since no automatic start of the auxiliary feedwater pumps, or auto alignment will occur for the deenergized trains. Safeguards actuation of auxiliary feedwater for Unit 2 is not required by TS while Unit 2 is defueled. An auto start of P-38A or P-38B on a Unit 1 safeguards auto signal is unaffected. No automatic functions of the operating unit are affected. This modification does not affect Unit 1 or Unit 2 auxiliary feedwater valve operations that are controlled by a Unit 1 safeguards signal. (SER 95-102)

## TEMPORARY MODIFICATIONS

The following tempoıary modifications were implemented as of the end of 1995:

1. TM 95-006, Containment Purge Supply and Exhaust. The TM supplies a temporary power source to several loads that lose normal power when molded-case circuit breakers are replaced in Unit 1 instrument buses during U1R22. TM 95-006 is installed prior to U1R22, with both units at power, while the Unit 1 containment purge supply and exhaust system is not in service. The temporary power supplies are in service for approximately 4 weeks. Unit 1 is in cold shutdown for most of the time the temporary modification is installed.

   Summary of Safety Evaluation: The following equipment is impacted by TM 95-006:

   - Containment isolation valves 1RM-3200A (1RE-211/212 monitor supply valve); 1RM-3200C (1RE-211/212 monitor return solenoid); 1VNPSE-3212 (1W6A&B purge exhaust fan suction valve), 1VNPSE-3244 (1W6A&B purge supply fan suction valve), 1RM-3200B (1RE-211/212 monitor supply valve), 1VNPSE-3213 (1W6A&B U1C purge exhaust fan suction valve), and 1VNPSE-3245 (1W-2A&B U1C purge supply van discharge valve).
   - RK-41 (Unit 1 containment purge system rack), RK-30 (PAB and SSB ventilation instrument rack), and RK-42 (drumming area HVAC equipment instrument rack).
   - PC-4012 and PC-4019, P-38A&B discharge valve pressure controllers.

   The safety function of the affected equipment is not degraded by the work. The failure modes of the affected equipment remain the same; the equipment will fail in the safe position upon a loss of power. In addition, the temporary power supplies have similar qualifications as the normal power supplies.

   The loss of power to 1RM-3200A-C causes a loss of 1RE-211 and 1RE-212, placing Unit 1 in a 48-hour LCO.

   The additional load added to each of the temporary power supplies (for MOB-048, MOB-049 and 1Y-05-13 loads) was evaluated to be no greater than 0.5 amps. This does not cause an overload on any of the temporary circuits.

   When power is lost to PC-4012, PC-4019, AF-4012 and AF-4019, the P-38A&B discharge air-operated valves (AOVs), go fully open. With those valves fully open, it is difficult to control auxiliary feedwater flow to the steam generators. To maintain power to the controllers while installing the temporary modification, temporary power is provided from another bus of the same qualifications. The additional loading was analyzed and determined to have no adverse effect on normal bus loading. Therefore, there is no adverse impact on plant safety resulting from the temporary power supply. (SER 95-043)

2. TM 95-019, Chemical and Volume Control. The TM replaces the 1CV-110C air-operated valve with a manual operator so the air operator for 1CV-110C can either be rebuilt or replaced.

Summary of Safety Evaluation: The manual operator for the 1CV-110C is normally maintained shut. 1CV-110C prevents dilution of the RCS with reactor makeup water flow to the top of the volume control tank (VCT). This valve is controlled via a 3-position switch on 1C-04 and opens automatically when dilute or alternate dilute is selected on the VCT makeup mode selector switch. By replacing the operator with a manual operator, these functions are disabled. During alternate dilute mode of operation, 1CV-110B also opens which allows the RCS dilution via this mode.

Manual borate, automatic makeup and manual blend modes are not affected because the flow path is through 1CV-110B. Dilute mode is not available from the Control Room without manual action through 1CV-110C. Alternate dilute is affected because 1CV-110C will not automatically open, but in alternate dilute, 1CV-110B also opens so an automatic flow path for dilution is established.

Operating Procedure OP-5B contains information for RCS hydrogen concentration. If a batch makeup of ≥2000 gallons or daily makeup of ≥4000 gallons is required and the entire amount is directed through 1CV-110B, then the flow will bypass the VCT and not pick up hydrogen gas from the VCT. This could result in the RCS hydrogen concentration to be reduced below the required concentration. An operator aid is placed at the Unit 1 makeup mode selector switch stating that the dilute mode is not available and in alternate dilute 1CV-110C does not open. If a batch dilution of ≥2000 gallons or a daily dilution of ≥4000 gallons is required, alternate dilute can be selected to establish dilution flow through 1CV-110B in accordance with OP-5B. 1CV-110C is then locally opened to establish a flow path to the top of the VCT and 1CV-110B is shut via the control switch at 1C-04 and the dilution operation is monitored. Prior to adding the entire amount, the CV-110B control switch is placed in automatic. The valve is verified open and 1CV-110C is then locally shut. This method of operation is only to be used for dilution and is only required if the batch or daily limits are reached. Blending operations to the RCS shall be in accordance with OP-5B with the flow path through 1CV-110B.

Since the only function disabled is the dilute mode of operation for VCT makeup and the flow path is still available via local operation if required for hydrogen control, installation of a manual operator on 1CV-110C does not increase the probability or consequences of an accident. There are no dilution alarms defeated by this change.

The manual valve materials are identical to the original air operator. Temperature, pressure and chemistry is not a concern with the change because the material is not changed. (SER 95-049)

3. TM 95-023, Waste Liquid. The TM replaces the 1WL-1728 operator and bonnet with a manual operator. The manual valve allows draining Sump A while the air operator is being rebuilt.

   Summary of Safety Evaluation: Containment isolation valve (CIV) WL-1728 is designed to isolate containment during an emergency. The upstream CIV WL-1723 is operational and controlled shut, except while draining Sump A. Failure of this valve is no more likely than failure of the valve with the air operator. The consequences of a failure is identical. WL-1723 upstream of WL-1728 provides the first isolation boundary during an accident. The unit shall not heat up past cold shutdown (RCS <200°F). This is controlled by containment closure checklist, CL-1E. (SER 95-054)

4. TM 95-024, Waste Liquid. The TM replaces the 1WL-1723 operator and bonnet with a manual operator and bonnet assembly. The manual valve allows draining of Sump A while the air operator is being rebuilt. The manual operator provides containment closure during cold or refueling shutdown conditions.

4. TM 95-024, Waste Liquid. The TM replaces the 1WL-1723 operator and bonnet with a manual operator and bonnet assembly. The manual valve allows draining of Sump A while the air operator is being rebuilt. The manual operator provides containment closure during cold or refueling shutdown conditions.

Summary of Safety Evaluation: 1WL-1723 is an air-operated diaphragm valve. The valve is controlled shut except while draining Sump A. Failure of this valve is no more likely than failure of the valve with an air operator. The consequences of a failure are identical. Unit 1 shall not heat up past cold shutdown (RCS < 200°F). This is controlled by containment closure checklist, CL-1E. The required time for closure of the system is the time-to-boil. (SER 95-055)

5. TM 95-034, 480 V Electrical. The deenergization of 2A-06 for MR 91-116*Y shuts down the various auxiliary systems required to maintain G-04 EDG in a standby condition; places portions of the emergency lighting system on battery power; and causes the EDG building fire detection system to be placed on battery power. Although G-04 is out of service while 2A-06 is out of service, power to the standby auxiliary systems is maintained during this time to expedite returning G-04 to service. TM 95-034 supplies temporary power to 2B-40 so these systems remain energized.

Summary of Safety Evaluation: During implementation of TM 95-034, G-03, G-04, and 2A-06 are out of service. An emergency power LCO is entered for 1A-06 and 2A-06 per TS 15.3.7. G-01 is aligned to 1A-05 and G-02 is aligned to 2A-05 to supply standby emergency power to the Unit 1 and 2 Train A safeguards equipment. Unit 2 is defueled with B-03 and B-04 crosstied during the implementation of TM 95-034. An evaluation shows that G-02 is not overloaded and verifies that 1B-40 can supply the additional loads required per TM 95-034.

The FSAR does not show that the EDGs or their associated support systems and connections cause or affect the probability of an accident. During the implementation of TM 95-034 neither G-03 or G-04 is relied upon for standby emergency power. G-01 and G-02 are available to provide standby emergency power to the Unit 1 and 2 Train A safeguards equipment. Therefore, implementation of TM 95-034 does not increase the probability of the occurrence of an accident or malfunction of equipment important to safety. (SER 95-109)

6. TM 95-037 (Unit 2), Primary Sampling. TM 95-037 installs an 8-channel ACR Smart Reader temperature logger. The logger has 7 remote temperature proves to record temperatures for 2POS-00955 and for conduit 2S735 near the delay coil. The data logger is located outside the missile shield to minimize its exposure to radiation.

Summary of Safety Evaluation: The data logger contains no aluminum. Therefore, it does not contribute to the generation of hydrogen as a result of the sodium hydroxide spray.

Since the case is resistant to fluids which it may be exposed to during an accident, it is concluded the data logger will maintain its structural integrity even following a LOCA. Therefore, it does not impact containment sump operation.

The high temperature probes are rated for 50°F to 335°F. The thermistor leads are Teflon insulated and jacketed. It is rated to 392°F. The conductors are silver coated copper. Teflon is water resistant and highly chemical resistant. The Teflon insulates and jackets the conductors. The thermistor lead wires will not be disturbed from installation to removal. The radiation dose is about 5x10 rads during normal operation. In the event of a LOCA, the Teflon may degrade further. If the Teflon comes off of the wiring

prevent undue clogging of the screen" Gibbs and Hill Evaluation "Paint and Insulation/Debris Effects on Containment Emergency Sump Performance" dated December 1989 identifies a maximum sump flow rate of 0.196 ft/sec. Therefore, the Teflon would not be transported to the sump.

The data logger is mounted to a pipe support, conduit support, or other mounting bracket. It is not attached directly to a pipe or conduit. Thermistor probes are attached to conduit 2S735 with Ty-raps. The leads from the thermistor are attached to the lighting conduit directly above 2S735, then routed as necessary along permanent conduit, component cooling water pipint, or mounting brackets through the opening in the missile shield wall to the lcoation selected for the data logger. A thermistor probe is also lcoated between the two Namco limit switches, 2POS-00955, on 2SC-00955. The lead for this thermistor may be routed along flexible conduit, rigid conduit, component cooling water piping, or mounting brackets outside the missile hsield to the data logger. The thermistor probes do not affect the operation of safety-related systems under any conditions. (SER 95-115)

## SPARE PARTS EQUIVALENCY EVALUATION DOCUMENTS (SPEEDS)

The following SPEEDs were implemented as of the end of 1995:

1. SPEED 94-057 Hi-Temp O-Rings for RCP Seals. The SPEED replaces several O-rings in the reactor coolant pump (RCP) seal package with O-rings designed for high temperature use. There is no change in form, fit or function. Only the specific EPDM compound of the replacement O-rings has changed.

   Summary of Safety Evaluation: The conversion to high temperature O-rings is recommended by Westinghouse as a method of minimizing concern with RCP seal O-ring failures. The high temperature O-ring compound was made available in 1989. The high temperature compound was used successfully by other utilities.

   For the RCP seals, the most harsh conditions are postulated from an extended loss of all ac to station auxiliaries. This would cause the loss of seal injection and component cooling thermal barrier cooling, allowing reactor coolant, at primary temperature and pressure, to pass through the seals. The extended loss of all ac power is not evaluated in the FSAR, but addressed by NRC Generic Safety Issue 23.

   The high temperature O-ring compound was specifically designed to withstand the conditions present during a loss of cooling to the RCP seals. The new material has superior tensile properties, compression set, and molding properties compared to those of the original compound. Extensive testing has shown that under high temperature conditions, (extended loss of all ac power), O-rings molded from the high temperature compound have a much greater probability of survival.

   The conversion to high temperature O-rings minimizes the impact of a loss of cooling to the RCP seals under accident conditions. There is no difference in the performance of the replacement O-rings compared to that of the original O-rings, either under normal operating conditions, or during any design basis accidents. (SER 95-031)

2. SPEED 94-071 and 94-072, 2" 600#/800# Velan Gate Valve Replaced with 2" Vogt Gate Valve. The SPEEDs replace the cavity cooler (1HX-30A&B) service water piping from service water (SW) supply header isolation valve SW-194 to containment penetrations P-37 and P-38. The replacement includes two 2" gate valves (1SW-203 and 1SW-205), two 3/4"gate valves (1SW-204 and 1SW-206), and two 2" piston check valves (1SW-30A&B).

   Summary of Safety Evaluation: In conjunction with replacement of the 2" piping between service water header valve 1SW-194 and containment penetrations P-37 and P-38, six valves are replaced. The piping is replaced because of external corrosion.

   The piping carries water from a service water supply header into containment to cavity coolers 1HX-30A&B. The work is performed during cold shutdown. SW supply header valve 1SW-194 and the cavity cooling system are isolated during the work. This is acceptable since cavity cooling is not needed during cold shutdown. Containment integrity is maintained by shutting cavity cooler inlet isolation valves 1SW-701A, 1SW-701B, 1SW-702A, and 1SW-702B which are inside containment. Valve 1SW-194 also

remains shut in order to isolate this piping from the south service water supply header. The replacement piping and valves are prefabricated in subassemblies in order to minimize the number of field welds required, and it expedites the work.

The replacement piping meets the Code requirements contained in FSAR Section 6.2-26 (USAS B31.1). All six replacement valves are constructed to meet ANSI B16.34 as required by FSAR Table 6.2-1 for valves 1SW-203 and 1SW-205. (SER 95-014).

## MISCELLANEOUS EVALUATIONS

The following evaluations were implemented as of the end of 1995:

1. BWNT Document #51-1179145-00. The document includes the design criteria and qualification testing performed on the B&W 0.875" rolled plug. An evaluation determined the ability of the plugs to serve as the primary-to-secondary pressure boundary in the steam generators. The effect of tube plugging on safe plant operation was assessed under the plugging procedure.

   Summary of Safety Evaluation: The B&W plugs are designed to meet design criteria outlined in FSAR Chapter 4, "Reactor Coolant System." The design and fabrication of the tube plugs is based on the requirements of the ASME Code Section III, Division 1, Subsection NB. The plugs are safety-related and are defined as ASME Section XI Class 1 components. Materials for the structural members of the plugs are in accordance with ASME Code Section III. Installation of the plugs shall be in accordance with ASME Code Section XI. Seismic classification of the roll plugs is in accordance with Regulatory Guide 1.29. The plugs are Seismic Class 1.

   The plugs are designed to provide a mechanical seal between the primary and secondary sides of the steam generator by lugging a tube end. Installation or removal of the plugs does not adversely affect the pressure retaining function or the structural integrity of the tube or tubesheet. The plug is designed so during installation no unacceptable residue or contaminant is left in the steam generator. The plug is compatible with primary and secondary water chemistry.

   The plugs are designed for a lifetime of 40 years. The leak rate from plugged tubes (based on 20% plugged) does not exceed 1 gallon per hour.

   The B&W 0.875" roll plug is suitable for installation in the steam generators because it meets the design criteria of the reactor coolant system pressure boundary as defined in the FSAR. Since the integrity of the RCS is not degraded in any way, safe operation is not affected. The qualification testing outlined in the referenced document was sufficient to determine that the plugs function as designed. (SER 95-047)

2. B&W Nuclear Technologies 51-129602-00. Re-rolling of Steam Generator Tubes Prior to Receiving a License Amendment. This document evaluates permissible plant configurations/modes given a completed re-roll repair with subsequent inspection of the re-roll, and concludes that proceeding to and maintaining hot shutdown prior to receiving approval for license amendment does not involve an unreviewed safety question.

   Summary of Safety Evaluation: The use of the re-roll repair technique to return Unit 2 to power requires a license amendment (Amendments 166/170). Tubes that have degradation exceeding the plugging limit were plugged or repaired via re-rolling. Re-rolled tubes were subsequently eddy current inspected in the tubesheet region. Re-rolled tubes not meeting the F* tube repair criterion were subsequently plugged.

Re-roll repair techniques are described in the B&W Nuclear Technologies report "W-44 F* Qualification Report" (BAW-10195 P, Revision 1, March 1994). The process provides a new hard roll which establishes a specified minimum distance F*. This assumes that the tube has a 360°, 100% through-wall crack at the indication. Therefore, no credit is given for the tube below the re-roll. The new hardroll is placed within the tubesheet, in an area free of degradation, above existing intra-tubesheet indications. The new hardroll has an undegraded length greater than or equal to the F* distance. The re-rolling process and application of the F* criterion maintains the tubes in service without reducing RCS flow. The re-roll becomes the new ASME Section III pressure boundary.

TS 15.4.2.A.6, "Corrective Measures," states that SG tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired prior to a return to power from a refueling or inservice inspection condition. The power conditions listed in TS 15.1 "Definitions" are "Power Operation" and "Low Power Operation." The TS definition of "Hot Shutdown" is the reactor is subcritical and $T_{avg}$ is ≥540°F. Hot shutdown is a shutdown condition and not a power condition, since the reactor is subcritical. Based on these two sections of TS, the evaluation and qualification of re-rolling for Unit 2 which demonstrate the structural integrity of the re-roll repair in accordance with Regulatory Guide 1.121 and ASME Section III requirements, and the verification via eddy current testing that all re-rolls in service meet the F* criteria, this safety evaluation evaluates whether bringing Unit 2 to a condition of hot shutdown, prior to receiving the approval of TS Change Request 184, involves an unreviewed safety question.

The structural integrity and leak limiting capability of the steam generator tubes after re-rolling and application of the F* criterion is equivalent to that of the original tubes. The F* criterion is equivalent to that of the original tubes. The F* criterion has been analyzed and tested for design, operating, and faulted condition loadings in accordance with Regulatory Guide 1.121 and ASME Code safety factors. The potential for tube rupture is not increased by re-rolling and application of the F* criterion. Resistance to tube rupture is strengthened by the presence of the tubesheet.

Hardrolling the tube into the tubesheet creates an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. The F* length of roll expansion is sufficient to preclude tube rollout due to tube degradation located below the F* distance, regardless of the extent of the tube degradation. Although unlikely, significant leakage from this region is fully bounded by the existing steam generator tube rupture analysis evaluated in the FSAR. In addition, primary-to-secondary leakage in excess of 500 gpd requires a controlled reactor shutdown and cooldown per TS 15.3.1.D.

The re-roll steam generator tube remains capable of performing its required heat transfer function. Re-rolling does not reduce the flow or provide barrier to heat transfer. Leaving a tube in service results in a more efficient steam generator and greater margin to the accident analysis limits than plugging an affected tube. The plugging percentages and protection setpoints remain within licensed limits.
(SER 95-112-01)

3. DCS 3.1.7. Service Water System Operability. To support maintenance and modification work on the non-essential service water motor operated valves (MOVs) (e.g., SW-2816, SW-2817, 1SW-2880, 2SW-2880, SW-2930A, and SW-2930B) a section was added to DCS 3.1.7 discussing the operability requirements of these valves. If the MOV is inoperable shut or isolated by its upstream manual valve, no LCO entry is required since the flow path has been secured. Based on the service water flow model, any one of these valves can be inoperable open and no LCO entry required as long as four service water (SW) pumps are operable and no other non-essential SW isolation valves are inoperable open. In addition, if SW-2816, SW-2930A or SW-2930B is inoperable open, the spent fuel pool valve (SW-2930A or SW-2930B) powered from the opposite train must be maintained shut. The above conditions ensure the automatic function of non-essential service water isolation is maintained. In addition to the above, in order to maintain the ability to isolate non-seismic piping downstream of an inoperable open non-essential service water MOV (e.g., all piping except for the seismic SFP return piping) either the MOV (e.g., SW-2816, SW-2817, 1SW-2880 or 2SW-2880) must be manually operable or the upstream manual valve must be operable. A Level 3 dedicated operator is then assigned to shut this valve in the case of a downstream pipe rupture as directed by the Control Room operator.

Summary of Safety Evaluation: The work does not change the consequences of an accident since the service water system's ability to provide the required amount of water to the essential loads is not impacted. This was validated by the service water (SW) WATER flow model. The non-essential service water MOVs, with the exception of the unaffected unit's turbine hall supply valve (1SW-2880 or 2SW-2880), receives an isolation signal if at least four service water pumps are not operating after a safety injection (SI) signal occurs and a 30-second time delay elapses after voltage is sensed on the safeguards buses.

The scenario that places the largest demand on the service water system is a LOCA in one unit combined with a loss of offsite power and failure of Train B emergency power. The WATER model assumes that only four SW pumps were initially operable. Therefore, the complete loss of Train B power results in only two operating SW pumps. This scenario results in a non-essential SW isolation signal which shuts only the MOVs powered by the Train A resulting in approximately 3380 gpm of unisolated Train B non-essential flow.

To ensure that the amount of unisolated non-essential SW does not exceed the amount assumed by the WATER model in the case of failure of Train A or B emergency power, four SW pumps must be operable with emergency power available, no other non-essential SW MOVs or AOVs (e.g., SW-LW-61 and SW-LW-62) may be inoperable open and the following actions are required:

IF SW-2816 is inoperable open, THEN SW-2930B must be maintained shut.
IF SW-2930A is inoperable open, THEN SW-2930B must be maintained shut.

IF SW-2817 is inoperable open, there are no other valve position requirements.
IF 1SW-2880 is inoperable open, there are no other valve position requirements.
IF 2SW-2880 is inoperable open, there are no other valve position requirements.
IF SW-2930B is inoperable open, THEN SW-2930A must be maintained shut.

The requirements ensure that the non-essential SW isolation function is maintained operable and does not require entry into an LCO for the inoperable MOV. It can also be noted that if any of the non-essential SW MOVs are inoperable shut that no LCO is required.

The work may involve removal of the valve operator, however this does not impact the pressure boundary or seismic adequacy of the SW piping. In the case of a leak or rupture of non-seismic piping downstream of SW-2916, SW-2817, 1SW-2880 or 2SW-2880, the assignment of a Level 3 dedicated operator to shut the MOV manually or to shut the upstream manual valve is adequate in addressing the flooding and spray concerns. Since the isolation of a pipe rupture downstream of a non-essential SW MOV normally requires operator action, the use of a dedicated operator to replace the remote control switch operation of the valve is considered justified. The isolation of a non-seismic piping rupture is not an immediate concern from a SW system cooling standpoint since the occurrence of a seismic event within 24 hours of a design basis accident is not credible.

The margin of safety as defined in the Technical Specifications is not reduced. Allowing one of the non-essential SW MOVs to be inoperable open as long as the above discussed actions are taken, meets the requirements of the Technical Specifications which requires that all necessary valves for the functioning of the SW system in an accident be operable. The system can provide the required flow to essential components with one of the MOVs inoperable as long as the specified conditions are met. (SER 95-032)

4. Revise Turbine Load Limit. Procedures exist, which include the turbine load limit when the crossover steam dump system becomes inoperable, that are revised in accordance with this evaluation. The current 480 MWe (gross) load limit specified in the procedures is reduced to 440 MWe. Letter WEP-95-5155 documents that a turbine load limit of 445 MWe (at an IOPS setting of 104% and condenser backpressure ≥0.8 in HgA) prevents turbine overspeed from exceeding the turbine's mechanical design limit (132% of rated speed) during a load rejection event, assuming no reliance on the crossover steam dump system for turbine overspeed protection. For conservatism, five additional megawatts were subtracted from the 445 MWe value to account for turbine load monitoring uncertainty to arrive at the 440 MWe limit. The 132% mechanical design limit for the turbine was established in 1970 by WCAP-7525-L and is discussed in FSAR Section 14.1.12.

Summary of Safety Evaluation: The need to reduce the current turbine load limit was identified via EWR 94-270 while investigating the origin of the 480 MWe value. Based on a 1974 curve of turbine load versus overspeed and recent confirming information, a load rejection from 480 MWe without crossover steam dump could potentially result in turbine overspeed reaching about 134% of rated speed. This exceeds the turbine's mechanical design limit of 132% overspeed.
Because the proposed change moves the turbine load limit in the conservative direction, the change does not create an unreviewed safety question because reducing the turbine load limit reduces the available energy in the turbine to generate a turbine missile. Therefore, the probability and consequences of generating a turbine are not increased by this change.

The probability of turbine damage and the offsite dose consequences resulting from a turbine missile impacting the spent fuel pool area are not increased. By reducing the turbine load limit, the likelihood of turbine damage is reduced, and the assumptions for generating a turbine missile at 132% of design speed are unaffected by this change. Therefore, the offsite dose consequences of a turbine missile, in the unlikely event that a missile would occur, remain the same as in the original analysis. (SER 95-058)

Summary of Safety Evaluation: To determine the correct load limit when crossover steam dump is not available, Westinghouse reviewed turbine overspeed data from 1970-1974. The resultant report provided a curve of turbine load as a function of condenser backpressure at an IOPS setting of 104% (the current plant setting). The curve conservatively assumes only turbine stop valve closure on an overspeed trip (no credit is taken for control valves closing). To account for possible monitoring instrument uncertainties, the Westinghouse curve has been adjusted to include maximum monitoring errors of 3 MW for load monitoring (±0.5% of span) and 0.04" HgA for condenser backpressure monitoring (±0.1% of span).

Because the proposed change moves the turbine load limit in the conservative direction, the change does not create an unreviewed safety question. Determination that there is not an unreviewed safety question as a result of this change is based on:

- Reducing the turbine load limit reduces the available energy in the turbine to generate a turbine missile. Therefore, the probability and consequences of generating a turbine missile (FSAR Section 14.1.12) are not increased by this change.

- The probability of turbine damage and the offsite dose consequences resulting from a turbine missile impacting the spent fuel pool area are not increased. By reducing the turbine load limit, the likelihood of turbine damage is reduced, and the assumptions for generating a turbine missile at 132% of design speed are unaffected by this change. Therefore, the offsite dose consequences of a turbine missile, in the unlikely event that a missile would occur, remain the same as in the original analysis.

- Reducing the turbine load limit does not create any new accident initiators or new equipment malfunctions not previously evaluated under FSAR Section 14.1.12. A more conservative turbine load limit has the opposite effect by creating conditions less likely to cause accidents and equipment malfunctions.

- No margin of safety defined in TS is affected by this change. By limiting turbine overspeed to 132% or less, the change does not affect the original assumptions in the turbine missile analysis. Therefore, the margin of safety for both turbine overspeed and offsite dose established in the original plant analysis is not affected by reducing the turbine load limit. (SER 95-058-01)

5. FSAR, Sections 11.1.4 and 11.2.5.

Summary of Safety Evaluation: The evaluation revision requires wording change for FSAR Section 11.1.4, Item 2 in the fourth paragraph. It currently states, "The gross activity monitor in the discharge line shall be operable." This should be replaced with the following words:

"If the gross activity monitor in the discharge line is not operable or if the discharge is made via pathway without an RMS monitor, the volume of liquid to be released is to be isotopically quantified pursuant to RETS 15.7.6 prior to release and periodically sampled during the release." (SER 94-053-01)

6. Regulatory Guide 1.97 Commitments. This evaluation supports removal of the boric acid storage tank (BAST) level indication from the list of parameters required to be monitored to meet commitments relative to Regulatory Guide 1.97 (RG 1.97) recommendations. This action involves a change in commitment only.

Summary of Safety Evaluation: RG 1.97 describes an acceptable method for providing instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. RG 1.97 identified five variable types (A, B, C, D, and E) to assist in selecting accident monitoring instrumentation and applicable criteria. Type D encompassed those variables that provide information to indicate the operation of individual safety systems important to safety. RG 1.97 also identified various design and qualification criteria which are delineated as Category 1, 2 and 3. Category 2 requires the instrumentation to meet environmental qualification standards, seismic qualification standards if part of a safety-related system, and be energized from a high-reliability power source. The method of display may be by dial, digital, CRT, or strip chart recorder.

RG 1.97, Table 2, identified the refueling water storage tank (RWST) level indication (to support the safety injection system) as a Type D, Category 2 variable. Boric acid storage tank (BAST) level indication was not identified in RG 1.97 as a post-accident monitoring variable. However, the WE September 1, 1983, response, BAST level was included along with RWST level as Type D, Category 2 variables for monitoring the operation of the safety injection (SI) system. This was included because at that time the BASTs were the initial suction source of SI. Since then, the design of the SI system has changed to use the RWST as the initial suction source of SI rather than the BASTs. Therefore, the BASTs are no longer part of the SI system flow path and BAST level indication is no longer required to monitor the operation of the SI system.

RG 1.97, Table 2, identified the volume control tank (VCT) level, charging flow, and letdown flow indications as Type D, Category 2 variables to monitor the operation of the chemical and volume control system (CVCS). In the WE September 1, 1983 letter, credit was also taken for BAST level as a variable to monitor the operation of the CVCS. However, Point Beach meets the RG 1.97 recommendations for monitoring CVCS operation without taking credit for BAST level indication, because charging flow (1&2FT-128), letdown flow (1&2FT-134), and VCT level (1&2LT-112&141) indications are maintained as Type D, Category 2 variables in accordance with the RG 1.97, Table 2 commitment. Therefore, this change meets the overall intent of RG 1.97 recommendations. (SER 95-006)

7. FSAR Section 9.3. The change increases boron concentration in the spent fuel pool from 2000 to 4000 ppm from the current concentration listed in the FSAR as 2000-2500 ppm.

Summary of Safety Evaluation: Boric acid concentration in the SFP is increased to satisfy a requirement of the independent spent fuel storage installation (ISFSI) Certificate of Compliance. The change assures that the multi-assembly sealed basket (MSB) cavity is filled with water having a boron concentration ≥2850 ppm. To ensure this requirement is met, boric acid concentration is increased above this level, up to a maximum of 4000 ppm boron.

The safety significance of having boron in the SFP is to ensure subcritical conditions during refueling. Changing the range of boron concentration requires a revision to FSAR Table 9.3-3. The FSAR changes include, where applicable, the fact that SFP boron and reactor cavity boron are no longer equal.

Increasing the SFP boric acid concentration is consistent with Westinghouse "Chemistry Criteria and Specifications," Table 1.7, "Specifications and Guidelines for Spent Fuel Pit Liquid," that indicates a nominal concentration of either 2000 or 4000 ppm boron, with an accompanying pH range, at 77°F, of 4.7 to 4.0. At 4000 ppm boron and 77°F, the expected pH of the SFP has been calculated by the CEQUIL program to be 4.22, which is within the range of Westinghouse guidelines.

Increasing the SFP boron concentration means that the SFP, the RWST and the refueling cavity, are at different boron concentrations. During refueling operations, when the SFP and cavity are connected, SFP boron can be diluted and cavity boron can be increased. Dilution of the SFP during periods when storage casks are not being loaded is not a safety concern, because the boron concentration in the SFP is increased from its present level. Similarly, increasing cavity boron above its present level during refueling operations is not a safety concern. It has been concluded that the RWST boron level should be kept at its current value. If the RWST boron concentration would be raised, a calculation must be performed showing what hot leg switchover times are now required and possibly changing the emergency operating procedure.

Therefore, the RWST boron level should be kept at less than 2500 ppm. After the SFP concentration is increased, it is possible the RWST level could exceed 2500 ppm at the end of a refueling. To ensure that RWST boron is less than 2500 ppm after refueling operations are complete, appropriate revisions to procedures CL-1D, CL-1A, and RP-1B were made. (SER 95-068)

8. FSAR Table 10.2-1, Carbohydrazide as a Steam Generator Wet Layup Chemical Treatment. Carbohydrazide (CHZ) is added to steam generator bulk water at a concentration of at least 10 ppm to provide increased protection against steam generator tube corrosion. This establishes a stronger reducing environment in the bulk water. FSAR Table 10.2-1, "AVT Control-Secondary Chemistry Control Guidelines," does not list a CHZ specification for use in wet layup as it does with hydrazine and ammonia.

Summary of Safety Evaluation: Wet layup chemicals are mentioned in the FSAR. The reason for the change from hydrazine to CHZ is one of increased personnel safety and performance at a reasonable cost. CHZ, [chemical formula of (N2H3)2CO], is an alternate dissolved oxygen scavenger designed to perform the same functions as hydrazine but at a faster rate and with less occupational environmental hazard. Other plants using CHZ have reported no problems.

Hydrazine is added to the wet layup bulk water treatment chemical in order to scavenge dissolved oxygen and provide a reducing environment to inhibit general and localized corrosion of ferrous materials (such as reducing pitting susceptibility of Alloy 600). EPRI and Westinghouse recommend a wet layup bulk water hydrazine concentration of at least 75 ppm with a bulk water pH of greater than 9.8 s.u. Bulk water pH is achieved by adding ammonia. Point Beach uses a catalyzed hydrazine product for wet layup of its steam generators since catalyzed hydrazine scavenges dissolved oxygen faster than hydrazine.

CHZ is a better iron passivator than hydrazine. It reacts with iron at a faster rate than hydrazine, catalyzed or not. CHZ is superior in its reaction rate with dissolved oxygen when compared to hydrazine, but slower when compared to catalyzed hydrazine. We obviously used catalyzed hydrazine for wet layup. Although CHZ reaction rate compared to catalyzed hydrazine is slower, it is not expected to be a concern. An evaluation shows there are no adverse affects when hydrazine is used instead of catalyzed hydrazine. To verify that the dissolved oxygen concentration in the wet layup water is acceptable, Chemistry analyzes the wet layup steam generator water for dissolved oxygen after the CHZ is added and mixed in the steam generators. If dissolved oxygen levels are found not to be acceptable, more CHZ is added. If dissolved oxygen is still unacceptable, catalyzed hydrazine is added to bring the dissolved oxygen concentration to normal. (SER 95-108)

9. FSAR Section 9.6.2, Service Water System. The FSAR section was revised to indicate that the highest expected service water inlet temperature is 75°F. No temperature reference currently exists in this section. The only reference to the maximum expected service water temperature is located in Section 9.3.1, "Spent Fuel Pool Cooling System." This section indicates that the highest expected service water inlet temperature is 60°F. This section is revised to include the 75°F temperature. Also the safety-related heat exchanger data tables are revised to include reference for operation with a service water inlet temperature of 75°F. The revisions are in response to service water vertical slice audit finding A-P-93-01 #4 and NRC SWSOPI Concern 93012-04.

Summary of Safety Evaluation: Calculations demonstrate that the service water system and associated safety-related heat exchangers have the capacity to meet their design basis heat transfer requirements at a service water inlet temperature up to and including 75°F. Adoption of a 75°F design temperature is within the operating capability of the service water system and safety-related heat exchangers and therefore does not increase the probability of occurrence or consequences of any accident evaluated in the FSAR. The probability of occurrence or consequences of a malfunction of equipment important to safety are not increased by adoption of the 75°F service water inlet temperature since this temperature is within the design temperature limits of the service water piping and the heat exchangers. Adequate heat transfer capability is maintained. Adoption of the 75°F design temperature does not create the possibility of an accident of a different type or create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FSAR since the required service water flows and heat transfer requirements for the safety-related heat exchangers are maintained. Since there is no change in the number of service water pumps required to be in operation or the number of safety-related heat exchangers required to transfer the required heat loads, the margin of safety defined in the Basis of TS 15.3.3 is not reduced (SER 95-042)

10. Offsite Dose Calculation Manual (ODCM), Appendix F. Under the NRC approved 10 CFR 20.2002 (formerly 10 CFR 20.302) exemption, contaminated sewage sludge disposal is permitted by land spreading on Company-owned fields in accordance with the approved restrictions and administrative controls. The controls specify the maximum concentration of radioactive materials in the sewage sludge, the annual volume of sewage sludge that may be disposed of on a per acre basis, and the maximum allowed doses to the maximally exposed individual and the inadvertent intruder. The disposal of contaminated sewage sludge is limited to an annual volume of 4000 gallons per acre. This limits the maximum total activity of contaminated sludge that can be spread on each acre each year and was proposed in response to an NRC request that ". . . WEPCO should commit to a maximum total activity of contaminated sludge to be spread per year." The intent is to control the dose that the farmer leasing the site receives while performing agricultural activities. The proposed change removes this administrative control because it is redundant to the dose control restrictions already in place in the exemption.

Summary of Safety Evaluation: The 4000 gallon per acre value is based on the assumption that the sewage sludge is contaminated with Co-58 at a concentration that is 10% of the 10 CFR 20 Appendix B, Table 2, Column 2 value. However, experience with past sewage sludge disposals has shown that the sludge may or may not be contaminated and, if contaminated, is at concentrations far below 10% of the 10 CFR 20 value.

An analysis expresses the 4000 gallon per acre annual limit in terms that more directly addresses the prior NRC request. Maximum cumulative activity limits on a per acre basis were calculated based on the maximally exposed individual and the inadvertent intruder annual dose limits for the radionuclides that were found in past sewage sludge disposals. These values were calculated using the methodology described the NRC approved 10 CFR 20.2002 exemption. This represents the amount of activity of each

radionuclide which gives the maximally exposed individual an annual dose of one millirem or the inadvertent intruder an annual dose of 5 millirem. Further review determined, however, that this limit is redundant to the dose calculation performed prior to each sewage sludge disposal and is not needed. Prior to each sewage sludge disposal, the dose to the maximally exposed individual and the inadvertent intruder is calculated to ensure that the doses caused by the contaminated sewage sludge are maintained within the dose limits of the 10 CFR 20.2002 exemption. This calculation sums the dose caused by the recent disposal and the dose caused by previous disposals on the disposal site. The calculation assumes that the disposal site size is 5 acres. For disposal sites less than 5 acres, the actual disposal size is used in the calculations. Therefore, for each disposal a de facto activity limit is imposed by the dose limit controls.

Removal of this restriction from the 10 CFR 20.2002 submittal does not reduce the margin of safety of our sewage sludge disposal activities nor reduces the commitment to keep doses resulting from sewage sludge disposal activities within the appropriate limits. (SER 95-057)

11. Re-Rolling of Steam Generator Tubes Prior to Receiving a License Amendment. Two repair strategies for Unit 2 steam generator tubes with repairable indications were pursued. One method is sleeving, the other is a re-rolling of the tube above the indication, higher up in the tube sheet area. The use of the re-roll repair technique to return the SG to service requires a license amendment. PBNP wants to begin the process of re-rolling prior to receiving the license amendment. This evaluation specifically covers only the actual re-rolling work while in cold shutdown. Because the re-roll repair process requires a license amendment, the tubes are not considered repaired in this evaluation and as such, Unit 2 is not able to leave cold shutdown or rely on the steam generators as a means of decay heat removal until tubes have been repaired or plugged by an approved method.

Summary of Safety Evaluation: Two trains of RHR are required to be operable if fuel is in the reactor and the cavity is not flooded. As a further restriction, pressurization of the RCS is not permitted (i.e., the RCS is maintained with an acceptable hot leg vent path) until further evaluations of the re-roll technique are completed. The results may be used to revise this evaluation to lift the no pressurization restriction while still keeping the unit in cold shutdown and maintaining two trains of RHR operable.

Re-rolling the tube does not preclude subsequent sleeve repair or plugging. Prior to raising RCS level to the point of flooding the SG channelheads (Unit 2 defueled with RCS at 3/4 pipe), a closeout inspection is performed.

The re-roll technique does not increase the probability or consequences of a previously analyzed accident or equipment malfunction or the possibility of ones of a different type. The unit remains in cold shutdown with an acceptable RCS hot leg vent path. The steam generators are out of service and decay heat removal (DHR) is provided by two operable trains of RHR if fuel is in the reactor and the cavity is not flooded. The re-roll does not cause failure of the tube. The re-roll is eddy current tested to verify a successful repair, thereby maintaining the boundary between primary and secondary which continues to serve as a barrier to dilution from the SG secondary side and as a containment boundary for shutdown safety. Re-rolls are verified successful or it is sleeved or plugged prior to loading the core. While in cold shutdown, the steam generators do not provide mitigation of a previously analyzed accident or equipment malfunction.

The refueling cavity or RCS may be filled provided the secondary side of the steam generator is intact up to the corresponding elevation in the refueling cavity or pressurizer. For example, if the cavity is filled, the steam generator secondary side must be intact up to at least El 66'. This prevents significantly draining the RCS or cavity assuming a failed tube. Maintaining an adequate hot leg vent path whenever fuel is in the

reactor prevents significant RCS pressurization given a complete loss of DHR with surge line flooding. WCAP-14089 Revision 1 shows that steam generator channel head pressures during a surge line flooding event remain acceptable for using nozzle dams given an acceptable hot leg vent path. It is unlikely that such low pressures could cause a tube failure. Even given tube failures on the cold leg, the hot leg vent path prevents the core from becoming uncovered.

TS 15.3.1.A requires two trains of RHR to be operable below 140°F. Between 140°F and 350°F two methods of DHR must be operable. The four possible methods given are the two steam generators and the two RHR loops. Before receiving a license amendment for the re-roll repair, the unit must remain in cold shutdown. Given the unlicensed repair method, TS 15.3.1.A must be met by having two trains of RHR operable if the cavity is not flooded, as the SG would be considered technically out of service. The two trains of RHR are acceptable for this TS and do not reduce the margin of safety. TS 15.3.1.D places limits on RCS leakage. Because the RCS remains depressurized, there is no driving head for leakage. Also, the TS directs that the unit be placed in cold shutdown if leakage limits are exceeded. The unit is maintained in cold shutdown. (SER 95-112)

12. License Amendment 157/161. The amendments revised TS 15.3.5, "Instrumentation System," and 15.4.1, "Operational Safety Review." This safety evaluation supplies a justification for procedure revisions made to implement Amendments 157 and 161.

    Summary of Safety Evaluation: The procedure changes made to implement the amendments do not involve an unreviewed safety question. The majority of the procedure changes incorporate the correct Technical Specification reference. The amendments change the numbering of many of the LCOs and surveillances. This requires the procedures to be revised with proper references. The amendments also have made existing surveillances into Technical Specification requirements. The associated procedures are revised solely to identify TS requirements.

    The remainder of Amendment 157/161 requirements are implemented by new procedures or new surveillances added to existing procedures. (SER 95-009)

13. License Amendment 158/162. The boric acid concentration in the boric acid storage tanks and CVCS system is reduced during U1R22 from a current nominal 12 wt% concentration to a new nominal concentration of 3.75 wt% (3.5-4.0 wt% control band). This change is in accordance with the new TS 15.3.2 that allows this operation.

    Summary of Safety Evaluation: Operation in the new control band of 3.5-4.0 wt% enables PBNP to maintain fluidity of the acid without the use or requirement of heat tracing. The TS minimum temperature for acid in this concentration range is 62.5°F. Use of this lower concentration boric acid results in a change in priority of the available methods of rapid boration and the quantities of boric acid that need to be added to compensate for stuck rods or an ATWS. This safety evaluation is to be used as a generic evaluation of changes to emergency/abnormal operating procedures. Procedure changes made need to be screened for applicability/compatibility with this safety evaluation.

The first type of change relates to the concern over use of unblended 12 wt% boric acid in non-heat traced lines of the CVCS and SI systems (including the use of emergency borate valve CV-350). The concern is that "freeze ups" will occur as the 12 wt% boric acid cools down as it makes its way down the pipe. A specific concern here is with the RCP seal injection lines and the RCP seals. Reducing the boric acid concentration to <4.0 wt% eliminates these concerns entirely because 4 wt% boric acid stays in solution above about 57.5°F. The procedures are changed to reflect the elimination of this concern.

The second type of change relates to the first type because the original operating practices stemmed from freeze-up concerns. The emergency borate line through CV-350 is designed for rapid boration of the RCS. This boration path utilizes 2" piping directly to the charging pump suction, whereas the blender path has only 1" piping. The achievable flow rate through CV-350 with one boric acid transfer pump in operation is in excess of 60 gpm, while the achievable flow through the blender is slightly less than 40 gpm. The emergency boration path is capable of about 60% more flow than the blender path. The higher boration flow combined with the elimination of freeze-up concerns makes the emergency boration path more desirable in situations where rapid boration is needed. Similar concerns with having the SI pumps take a suction directly from the BASTs, if desired, are also addressed. The procedure changes reflect these types of changes in capability/priority.

The third type of change relates to the amounts of boric acid required for certain functions. Because of the lower concentration, a larger volume of low concentration acid is required. An example of this is the amount of boric acid needed to be added to compensate for stuck rods. The net effect is the same for these cases whether low or high concentration acid is utilized; a certain amount of negative reactivity is added to the RCS. Other means of boration if the BASTs are not available are the CVCS paths from the RWST and the safety-related means of boration, depressurize and use of SI from the RWST.

The other types of changes relate to the de-energization and isolation of the boric acid heat tracing after the concentration is reduced. The only loads remaining on boric acid heat tracing circuits HTPA and HTPB are a few facade freeze protection circuits and recorder power. HTPA is stripped on an SI signal. The reported loadings on these panels are updated.

The changes to the procedures do not change the intent of the procedures; they reflect the use of lower concentration boric acid to accomplish the same goals. The CVCS system is still capable of performing its boration function. The systems and equipment are still used in accordance with their design functions and reliability should increase with the lower concentrations. (SER 95-045)

14. Unit 1 Cycle 23 Core Reload. The U1C23 core loading pattern is described in letter 95WE*-G-0001, dated January 17, 1995. This safety evaluation covers the mechanical design, nuclear design, thermal-hydraulic design, power capability, FSAR accidents and TS changes that apply to the U1C23 reactor core.

Summary of Safety Evaluation: The Westinghouse U1C23 Reload Safety Analysis Checklist (RSAC) was reviewed. The applicable limiting values for Cycle 23 are consistent with those in previous safety evaluations. An exception is the boron concentration in the RWST. Westinghouse evaluates the boron concentration in the containment sump following a LOCA for every reload core. The U1C23 evaluation shows that the core may return to a critical condition following a LOCA with a boron concentration of 2000 ppm in the RWST. The Westinghouse evaluation includes an uncertainty of 100 ppm. Administratively controlling the RWST boron concentration to at least 2100 ppm prevents a post-LOCA return to criticality for U1C23 with additional margin for future cycles. The administrative control changes

the minimum RWST boron concentration allowed in procedure NP 3.2.2 from 2000 to 2100 ppm. Procedure EOP 1.3 is also changed to ensure RWST inventory is delivered to the RCS or containment sump to an indicated level of 28% for containment sump recirculation within 8 hours after shutdown.

As a result of the U1C23 evaluation, it is concluded that its design does not cause safety limits to be exceeded, provided that the following conditions are met:

- U1C22 burnup is bounded by 10,000 and 11,000 MWD/MTU. (Actual U1C22 burnup was 10,848 MWD/MTU.)
- U1C23 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 U1C23 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 Technical Specifications.
- Administrative controls ensure the boron concentration in the Unit 1 RWST is at least 2100 ppm.

Regions 25A and 25B fuel assemblies are Westinghouse upgraded OFA, and have the same mechanical design as the previous Regions 24A and 24B upgraded OFAs. Thimble plugs are not used in the Cycle 23 core.

A justification for continued operation (JCO) supports operation at a nominal reactor vessel $T_{avg}$ of 570.0°F. Most accident analyses assume a nominal vessel $T_{avg}$ of 573.9°F. Systems and components analysis was performed at a variety of temperatures which may not bound operation at 570°F. The basis for this JCO was reviewed by Westinghouse and concluded that the JCO remains valid for U1C23.

Operation of the U1C23 core does not involve an increase in the probability or consequences of accidents previously considered, does not involve a decrease in safety margin, and does not involve a significant hazard consideration. Therefore, provided that the startup physics testing does not result in discrepancies with the analysis assumptions, the operation of U1C23 in accordance with TS is acceptable based on the reload design and this safety evaluation. (SER 95-050)

15. Unit 2 Cycle 22 Reload. This safety evaluation covers the mechanical design, nuclear design, thermal-hydraulic design, power capability, FSAR accidents, and TS changes that apply to the U2C22 reactor core.

Summary of Safety Evaluation: The Westinghouse Unit 2 Cycle 22 Reload Safety Analysis Checklist (RSAC) was reviewed. The applicable limiting values for Cycle 22 are consistent with those used in previous safety evaluations. As a result of the U2C22 evaluation, it is concluded that its design does not cause safety limits to be exceeded, provided that the following conditions are met:

- U2C21 burnup is bounded by 10,800 and 11,700 MWD/MTU: (Actual U2C21 burnup was 11,226 MWD/MTU).

- U2C22 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 ≤10 ppm of boric acid at the U2C22 rated power condition of 1518.5 MWt, plus 1500 MWD/MTU power costdown operation.

- There is adherence to the plant operating limitations given in TS.

- Administrative controls are in place to ensure the boron concentration in the Unit 2 RWST is at least 2100 ppm.

- Total control rod worth uncertainty is 7% instead of 10% in plant procedures.

Regions 24A, 24B, 24C, and 24D fuel assemblies are Westinghouse upgraded OFA, and have the same mechanical design as the previous Region 23A, 23B, 23C, and 23D upgraded OFAs except for the following fuel design improvement. Thimble plugs are not used in the U2C22 core.

The integral fuel burnable absorbers (IFBAs) and axial blankets described in WCAP-11872 are incorporated in U2C22. IFBAs have been incorporated in Regions 24A, 24B, and 24C. Four fuel assemblies in Region 24A contain 8 IFBA rods, 9 fuel assemblies in Region 24B contain 32 IFBA rods, and 8 fuel assemblies in Region 24C contain 16 IFBA rods. Fuel in Regions 24A, 24B, 24C, and 24D contain axial blankets.

Steam generator A has a higher anticipated tube plugging percentage (19.4%) than steam generator B (16.6%). The asymmetry of the tube plugging above the average (18%) has been assessed by Westinghouse and the accident analyses remain valid.

The design basis that the core remain subcritical on soluble boron alone in long-term cooling following a large break LOCA is satisfied for U2C22 operation based on a minimum refueling water storage tank (RWST) boron concentration of 2100 ppm. Westinghouse evaluates the boron concentration in the containment sump following a LOCA for every reload core. The evaluation for U1C23 showed that the core could return to a critical condition following a LOCA with a boron concentration of 2000 ppm in the RWST. A boron concentration of at least 2100 ppm prevents a post-LOCA return to criticality for U1C23 with additional margin for future cycles. An administrative control was implemented for U1C23 by changing the minimum RWST boron concentration allowed in procedures from 2000 to 2100 ppm. This remains in effect for U2C22. (SER 95-111)

Summary of Safety Evaluation: The evaluation revision adds the following conditions to be met:

- Prior to taking the reactor critical, Technical Specification Change Request 181 is approved by the NRC, if the minimum raw measured total flow rate is less than 174,000 gpm.

- Prior to taking the reactor critical, Technical Specification Change Request 184 is approved by the NRC for steam generator tube repair.

Steam generator A has a higher anticipated tube plugging percentage (25.6%) than steam generator B (19.0%). The asymmetry of the tube plugging about the average (22.3%) has been assessed and the analyses remain valid. LOCA analyses remain valid for asymmetric tube plugging as long as the plugging level in any steam generator remains less than or equal to 30%, the total thermal design flow rate of 166,000 gpm is maintained, and neither loop is less than 95% of the average loop thermal design flow.

Non-LOCA analyses are unaffected by the asymmetric tube plugging provided the reactor protection system setpoints remain within design limits. Procedures are in place to maintain RPS setpoints within design limits by adjusting the indicated ΔTs to be conservative when compared to actual plant values. Systems and components analyses remain valid for asymmetric tube plugging as long as the steam pressure in the steam generators remains above 450 psia and the thermal load of either steam generator remains below 825 MWt. (SER 95-111-01)

16. Work Orders. The work orders (WOs) replace breakers in safety-related instrument panels 1Y-01, 1Y-02, 1Y-03, 1Y-04, non-safety-related panels 1Y-05 and 1Y-06, and breakers with connected loads in instrument panels 1Y-11, 1Y-21, 1Y-31, and 1Y-41. Also, the breakers in 1Y-11, 1Y-21, 1Y-31, 1Y-41, 1Y-101, 1Y-102, and downstream breakers fed by these panels are cycled.

Summary of Safety Evaluation: An extensive review of work orders including verification of sources, walkdowns of actual plant equipment, and walkdowns of the process was performed. The work occurs during a defueled condition so impact on the plant is minimal.

For equipment requiring power during the breaker replacement, temporary modifications are performed to supply temporary power. The following equipment has power supplied using temporary modifications for the duration of the instrument bus work: Unit 1 containment purge supply and exhaust controls, P-38A&B discharge valve pressure controllers (PC-4012 and PC-4019), fire protection panel D-400, MOB-047 that provides control power to various PAB ventilation fans, Control Room and computer room air conditioning panels C67/C58, and turbine electrohydraulic control panel 1C-39.

Previous experience and lessons learned from the breaker replacement in Unit 2 instrument panels were incorporated into Unit 1 work orders. Implementation of new inserts in panels 1Y-01, 1Y-02, 1Y-03, 1Y-04, 1Y-05 and 1Y-06 are utilized as improvements. This allows the breakers to be prestaged on the new inserts, which expedites the breaker replacements to minimize the time the instrument buses are out of service.

Each panel is removed from service for approximately 12 hours to perform the breaker replacement. Removal of the instrument buses from service does not result in a trip of Unit 2, or cause automatic initiation of equipment associated with the operation of Unit 2. The equipment was reviewed to determine which has a potential impact on either the defueled Unit 1, the operating Unit 2 or the spent fuel pool. Equipment having temporary power supplied during the breaker replacement was determined to have no impact on either unit or the spent fuel pool.

Equipment required to mitigate an accident is not affected. Neither set of redundant instrument buses is taken out of service at a time. Prior to securing power on the buses, the associated indication circuit on the other redundant bus is verified operational. Since this work is performed during the defueled time for Unit 1, the equipment necessary to mitigate an accident is not required to be in service. Most of the equipment required to mitigate an accident on Unit 2, or the spent fuel pool is not affected by the instrument bus work. The only exception is the motor-driven auxiliary feedwater pump P-38A, which is out of service for the emergency diesel generator addition. Therefore, the probability of a malfunction of equipment important to safety is not increased. (SER 95-048)

17. WOs 9505520 and 9505521, EDG Heat Exchanger Endbell Epoxy. During the G-02 EDG outage to facilitate realigning Train B to Train A, an epoxy coating is applied to the HX-55B-1 and HX-55B-2 heat exchanger endbells. Application of this epoxy coating mitigates corrosion and ensures the reliability of the heat exchangers and the G-02 EDG.

Summary of Safety Evaluation: Interior surfaces of the heat exchanger endbells are prepared for coating by abrasive blasting, and the new coating material applied is a 100% solids non-solvent emitting epoxy resin. One to two coats of Duromar SAR epoxy is applied to the heat exchanger divider plate, for a total coating thickness of approximately 0.050". The remaining interior surfaces of each endbell are coated with one coat of Plastocor 2000U epoxy with a minimum thickness of approximately 0.040" sandwiched between two coats of Plastocor 400U epoxy each with a thickness of 0.010 to 0.020". Endbell flange faces are coated with 0.015 to 0.020" of Plastocor 400U epoxy.

The coating process provides an excellent long-term protective layer which adheres tightly to the base metal and does not affect the operation of the heat exchanger. This type epoxy coating was applied to component cooling water heat exchangers. After approximately a year of service, no adverse effects to the heat exchangers or degradation of the coatings have been detected.

Since the coatings are only applied to the heat exchanger endbell surfaces, the heat transfer characteristics and operability of the heat exchangers, and therefore the operability of G-02 is not affected. The epoxy coating is designed to maintain a tight, cohesive bond to the base metal to which it is applied in a service water environment up to temperatures of 220°F. (SER 95-063)

## NUMBER OF PERSONNEL AND PERSON-REM BY WORK GROUP AND JOB FUNCTION - 1995

| Job Group Station Employees | Number of Personnel Greater Than 100 mrem | Total rem for Job Group | Work Function and Total Person-rem | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | Reactor Operations & Surveillance | Routine Maintenance | Inspections | Special Maintenance | Waste Processing | Refueling |
| Operations | 55 | 13.260 | 6.540 | ------- | 4.700 | ------- | 0.150 | 1.870 |
| Maintenance | 44 | 43.180 | ------- | 24.950 | 0.330 | 3.860 | ------- | 14.040 |
| Chemistry & Health Physics | 27 | 12.000 | 11.550 | ------- | ------- | ------- | 0.450 | ------- |
| Instrumentation & Control | 12 | 2.440 | ------- | 1.660 | 0.040 | ------- | ------- | 0.740 |
| Administration & Engineering, Regulatory Services | 8 | 2.560 | 0.090 | ------- | 2.470 | ------- | ------- | ------- |
| Utility Employees | 50 | 24.680 | 1.240 | 21.880 | 1.560 | ------- | ------- | ------- |
| Contractor Workers & Others | 190 | 91.919 | 0.400 | ------- | 9.900 | 79.499 | 2.120 | ------- |
| GRAND TOTALS | 386 | 190.039 | 19.820 | 48.490 | 19.000 | 83.359 | 2.720 | 16.650 |

1450 individuals were monitored exempt from the provisions of 10 CFR 20.

## VI. STEAM GENERATOR EDDY CURRENT TESTING

The following abbreviations are used throughout the rest of the report.

| LIST OF ABBREVIATIONS | | | |
|---|---|---|---|
| % | Percent Through Wall Indication | DI | Distorted Indications |
| MAI | Multiple Axial Indication | FL | Full Length |
| SAI | Single Axial Indications | TEC/H | Tube End C/H Leg |
| NQI | Non-quantifiable Indications | STC/H | Sleeve Top C/H Leg |
| PTF | Parent Tube Flaw | TSC/H | Tubesheet C/H Leg |
| DRI | Distorted Roll Ind | BUH | Bottom of Upper HEJ Joint (HL) |
| HRH | Bottom of Hardroll (HL) | SCI | Single Circumferential Ind |
| SVI | Single Volumetric Indication | 01-06H or C | Tube Support Plate No. Hot or Cold Leg |
| BPH or C | Baffle Plate Hot or Cold Leg | AV1-4 | Anti-vibration bar number |
| TSH or C | Tubesheet Hot or Cold Leg | | |

## UNIT 1

Inspection Plan: During the Unit 1 Refueling 22 outage, eddy current testing was performed March 21, 1995 to March 25, 1995. Full length eddy current testing was performed on 100% of the tubes in each steam generator. The extent tested in each steam generator is as follows:

| Eddy Current Inspection Plan | | |
|---|---|---|
| **Extent of Inspection** | **Number of Tubes** | |
| | **A SG Hot/(Cold)** | **B SG Hot/(Cold)** |
| Unsleeved Tubes: Full Length | 3210 | 3210 |
| RPC: Length of Tubesheet | 21 | 20 |
| **TOTALS** | 3231/(0) | 3230/(0) |

Inspection Results: The results of these inspections revealed 6 tubes in the A steam generator with reportable indications, and 5 in the B steam generator. The following is a summary of the eddy current inspections results listing the largest reportable indication per tube:

| Eddy Current Inspection Results Hot Leg (Cold Leg) | | |
|---|---|---|
| | A SG | B SG |
| DI | 0 (0) | 0 (0) |
| DRI | 0 (0) | 0 (0) |
| 20-29% | 2 (2) | 4 (0) |
| 30-39% | 1 (1) | 0 (0) |
| 40-49% | 0 (0) | 1 (0) |
| ≥50% | 0 (0) | 0 (0) |
| NQI | 0 (0) | 0 (0) |
| Axial Ind. | 0 (0) | 0 (0) |
| Sleeve Ind. | 0 (0) | 0 (0) |
| **TOTALS** | 3 (3) | 5 (0) |

## A Steam Generator Indications

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

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 15-49 | 33 | 04H | 30.0 |
| 29-40 | 20 | AV2 | 0.8 |
| 40-42 | 23 | AV1 | 0.6 |
| 10-45 | 24 | 03C | 20.9 |
| 16-89 | 34 | TSC | 15.8 |
| 16-89 | 26 | TSC | 16.9 |
| 22-47 | 20 | TSC | 21.4 |
| 22-47 | 24 | TSC | 20.4 |

## B Steam Generator Indications

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

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 32-38 | 23 | AV3 | 0.1 |
| 32-38 | 23 | AV4 | 0.0 |
| 35-51 | 22 | AV1 | 0.0 |
| 35-51 | 23 | AV3 | 0.0 |
| 35-51 | 45 | AV2 | 0.1 |
| 37-68 | 25 | 05H | 32.7 |
| 37-69 | 24 | 05H | 12.3 |
| 39-67 | 26 | 06H | 1.3 |

Repaired or Plugged Tubes: Plugging was not required in the A steam generator. One was plugged because it exceeded the Technical Specification plugging limit of 40%.

| TUBES PLUGGED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R35C51 | 45% | AV2 0.1 |

## UNIT 2

Inspection Plan: During the Unit 2 Refueling 21 outage, eddy current testing was performed October 16, 1995 to October 25, 1995. Full length eddy current testing was performed on 100% of the unsleeved tubes in each steam generator. In addition, 20% of the sleeved tubes were inspected and 20% of the cold leg unsleeved tubes not included in any inspection plan were inspected over the unsleeved length. The extent tested in each steam generator is as follows:

| Eddy Current Inspection Plan | | |
|---|---|---|
| **Extent of Inspection** | **Number of Tubes** | |
| | **A SG Hot/(Cold)** | **B SG Hot/(Cold)** |
| Unsleeved Tubes: Full Length | 1513 | 1516 |
| Sleeved Tubes: Unsleeved Length | (314) | (732) |
| Sleeves | 1343/(24) | 1330/(141) |
| RPC: Length of Tubesheet | 1411 | 1547 |
| **TOTALS** | 4267/(338) | 4393/(873) |

**NOTE**: As a result of the preliminary results from the 20% sleeve inspection, a decision was made to expand to 100% of the hot leg sleeves based on a Technical Specification C-3 categorization. There was no need to expand on the cold leg sleeve program.

Inspection Results: The results of these inspections revealed 576 tubes in the A steam generator with reportable indications, and 402 in the B steam generator. The following is a summary of the eddy current inspection results listing the largest reportable indication per tube:

| Eddy Current Inspection Results Hot Leg (Cold Leg) | | |
|---|---|---|
| | **A SG** | **B SG** |
| 20-29% | 2 (3) | 16 (32) |
| 30-39% | 2 (3) | 7 (25) |
| 40-49% | 4 (0) | 0 (2) |
| ≥50% | 0 (1) | 0 (0) |
| NQI | 0 (0) | 0 (0) |
| SVI | 12 (0) | 3 (2) |
| Axial Ind | 350 (0) | 248 (0) |
| Circ Ind | 1 (0) | 0 (0) |
| Sleeve Ind | 198 (0) | 67 (0) |
| **TOTALS** | 569 (7) | 341 (61) |

## A Steam Generator Indications

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

The following indications are reportable but do not exceed the repair limit.

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 16-5 | 38 | 02H | 0.2 |
| 2-13 | 22 | 01H | 34.1 |
| 1-18 | 31 | 01H | 18.6 |
| 1-18 | 36 | 01H | 18.9 |
| 1-20 | 37 | 01H | 18.7 |
| 1-24 | 26 | 01H | 27.1 |
| 40-26 | 23 | TSH | 4.1 |
| 40-26 | 34 | TSH | 5.9 |
| 34-39 | 22 | TEH | 3.1 |
| 43-50 | 33 | 01H | 0.5 |
| 42-51 | 28 | TSH | 0.4 |
| 45-52 | 22 | 02H | 0.2 |
| 7-78 | 21 | TSH | 0.8 |
| 14-26 | 25 | TSC | 0.6 |
| 4-31 | 28 | TSC | 0.9 |
| 4-31 | 35 | TSC | 0.7 |
| 8-36 | 27 | TSC | 1.2 |
| 14-48 | 27 | TSC | 1.0 |
| 39-50 | 32 | 06C | -0.1 |
| 9-61 | 35 | TSC | 0.2 |
| 1-91 | 35 | 01C | -0.3 |

## B Steam Generator Indications

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

The following indications are reportable but do not exceed the repair limit.

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 7-1 | 29 | 01H | 44.0 |
| 6-15 | 34 | TSH | 43.1 |
| 6-16 | 29 | TSH | 0.2 |
| 28-16 | 27 | TSH | 45.6 |
| 28-16 | 29 | TSH | 47.2 |
| 22-24 | 35 | 01H | -23.1 |
| 22-26 | 21 | 01H | -12.4 |
| 22-28 | 24 | 01H | 0.2 |
| 22-28 | 26 | 01H | 11.6 |
| 29-30 | 25 | 01H | 11.5 |
| 27-31 | 22 | 01H | 13.5 |
| 32-38 | 26 | 01H | -9.4 |
| 34-39 | 21 | 01H | 7.9 |

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 45-47 | 38 | 02H | 0.3 |
| 33-48 | 34 | TSH | 0.5 |
| 43-58 | 31 | TSH | 2.3 |
| 42-60 | 27 | TSH | 0.7 |
| 42-60 | 23 | TSH | 6.0 |
| 43-60 | 24 | TSH | 2.6 |
| 26-68 | 31 | 01H | 9.3 |
| 6-73 | 25 | 01H | -11.8 |
| 27-79 | 24 | TSH | 3.8 |
| 28-79 | 22 | TSH | 5.7 |
| 9-81 | 26 | 01H | 19.4 |
| 3-82 | 31 | 05H | -0.4 |
| 12-89 | 26 | TSH | 2.9 |
| 8-91 | 26 | TSH | 41.9 |
| 8-91 | 21 | TSH | 49.6 |
| 1-2 | 26 | 01C | -0.2 |
| 5-2 | 27 | 01C | -0.1 |
| 12-2 | 35 | 01C | 0.2 |
| 20-6 | 21 | 01C | 0.0 |
| 22-8 | 28 | 01C | -0.3 |
| 33-19 | 28 | 01C | -0.2 |
| 36-21 | 20 | 01C | -0.1 |
| 37-21 | 36 | 01C | -0.2 |
| 34-22 | 25 | 01C | -0.2 |
| 35-22 | 36 | TSC | 12.2 |
| 34-23 | 24 | 01C | -0.2 |
| 37-23 | 34 | 01C | -0.2 |
| 26-24 | 31 | 01C | -0.2 |
| 34-24 | 29 | 01C | -0.2 |
| 34-25 | 29 | 01C | -0.2 |
| 36-25 | 25 | 01C | -0.2 |
| 39-25 | 27 | 01C | -0.3 |
| 37-27 | 32 | 01C | -0.2 |
| 39-27 | 38 | 01C | -0.1 |
| 29-28 | 26 | 01C | -0.3 |
| 31-28 | 26 | 01C | -0.2 |
| 27-30 | 32 | 01C | -0.2 |
| 29-30 | 22 | 01C | -0.3 |
| 36-31 | 30 | 01C | -0.1 |
| 32-32 | 29 | 01C | -0.2 |
| 31-36 | 33 | 01C | -0.2 |
| 28-39 | 33 | 01C | -0.2 |
| 34-39 | 33 | 01C | -0.1 |
| 29-41 | 26 | 01C | 0.1 |
| 37-42 | 30 | 01C | -0.3 |
| 33-46 | 30 | 01C | -0.1 |
| 30-47 | 25 | 01C | 0.0 |
| 30-48 | 35 | 01C | -0.1 |
| 31-48 | 31 | 01C | 0.0 |

| Row - Column | Indication | Location | Inch Mark |
|---|---|---|---|
| 33-48 | 34 | 01C | 0.0 |
| 38-48 | 24 | 01C | -0.4 |
| 41-48 | 21 | 01C | -0.2 |
| 30-49 | 21 | 01C | -0.1 |
| 38-49 | 21 | 01C | -0.1 |
| 42-49 | 30 | 01C | -0.2 |
| 27-51 | 39 | 01C | -0.2 |
| 38-52 | 32 | 01C | -0.3 |
| 32-53 | 25 | 01C | -0.2 |
| 38-53 | 20 | 01C | -0.2 |
| 38-54 | 30 | 01C | -0.2 |
| 38-56 | 24 | 01C | -0.2 |
| 20-58 | 27 | TSC | 0.8 |
| 33-58 | 28 | 01C | -0.2 |
| 15-62 | 25 | TSC | 1.1 |
| 36-63 | 25 | 01C | -0.2 |
| 37-63 | 24 | 01C | -0.1 |
| 32-65 | 20 | 01C | -0.2 |
| 36-65 | 26 | 01C | -0.3 |
| 31-66 | 38 | 01C | 0.1 |
| 3-67 | 23 | TSC | 1.6 |
| 31-67 | 25 | 01C | 0.0 |
| 37-67 | 20 | 01C | -0.2 |
| 3-69 | 20 | TSC | 0.8 |
| 4-71 | 27 | TSC | 0.7 |
| 13-71 | 23 | TSC | 0.4 |
| 33-73 | 33 | 01C | -0.2 |
| 35-44 | 27 | AV2 | 0.1 |
| 28-61 | 37 | AV3 | 16.3 |
| 27-62 | 31 | AV3 | 15.8 |
| 28-62 | 38 | AV3 | 16.4 |

Repaired or Plugged Tubes: Tube repairs were performed on each steam generator during U2R21 as a result of eddy current indications. Tubes that exceeded the Technical Specification repair limit were either plugged or repaired using the re-rolling process and application of the F* criteria. There were 244 tubes plugged in the A steam generator and 117 tubes plugged in the B steam generator. Of the 117 tubes plugged in B, 4 were plugged because re-rolls were performed incorrectly. There were 323 tubes repaired by re-roll in the A steam generator and 209 tubes repaired in the B steam generator. The increase in the number of tubes requiring repair is attributed to enhancements in the eddy current technology used. This was the first time the Plus Point probe was used.

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R15C20 | PTF | BUH 2.0 |
| R9C21 | PTF | BUH 2.2 |
| R5C23 | PTF | BUH 2.5 |
| R11C23 | PTF | BUH 2.2 |
| R12C23 | PTF | BUH 2.4 |
| R18C23 | PTF | BUH 2.4 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| **TUBE** | **DEFECT** | **LOCATION** |
| R8C24 | PTF | BUH 2.2 |
| R12C24 | PTF | BUH 2.4 |
| R2C25 | PTF | BUH 2.2 |
| R3C26 | PTF | BUH 2.2 |
| R4C26 | PTF | BUH 2.4 |
| R5C26 | PTF | BUH 2.3 |
| R12C26 | PTF | BUH 2.3 |
| R9C27 | PTF | BUH 2.0 |
| R10C27 | PTF | BUH 1.6 |
| R11C27 | PTF | BUH 2.2 |
| R12C27 | PTF | BUH 2.5 |
| R13C27 | PTF | BUH 2.3 |
| R2[illegible]C27 | PTF | BUH 2.3 |
| [illegible]5C28 | PTF | BUH 2.3 |
| [illegible]3C29 | PTF | BUH 2.3 |
| [illegible]C29 | PTF | BUH 2.0 |
| R11C29 | PTF | BUH 2.2 |
| R3C30 | PTF | BUH 2.1 |
| R4C30 | PTF | BUH 2.5 |
| R5C30 | PTF | BUH 2.2 |
| R8C30 | PTF | BUH 1.0 |
| R8C30 | PTF | BUH 1.9 |
| R9C30 | PTF | BUH 2.0 |
| R12C30 | PTF | BUH 2.0 |
| R18C30 | PTF | BUH 2.3 |
| R23C30 | PTF | BUH 2.2 |
| R26C30 | PTF | BUH 2.1 |
| R3C31 | PTF | BUH 2.0 |
| R5C31 | PTF | BUH 2.0 |
| R6C31 | PTF | BUH 2.1 |
| R8C31 | PTF | BUH 2.5 |
| R9C31 | PTF | BUH 2.3 |
| R10C31 | PTF | BUH 2.3 |
| R16C31 | PTF | BUH 1.9 |
| R18C31 | PTF | BUH 2.2 |
| R20C31 | PTF | BUH 2.0 |
| R3C32 | PTF | BUH 2.3 |
| R5C32 | PTF | BUH 2.3 |
| R10C32 | PTF | BUH 2.1 |
| R15C32 | PTF | BUH 2.4 |
| R20C32 | PTF | BUH 2.2 |
| R3C33 | PTF | BUH 2.3 |
| R5C33 | PTF | BUH 2.3 |
| R7C33 | PTF | BUH 2.0 |
| R13C33 | PTF | BUH 2.4 |
| R14C33 | PTF | BUH 1.7 |
| R6C34 | PTF | BUH 2.5 |
| R3C35 | PTF | BUH 2.4 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R18C35 | PTF | BUH 0.8 |
| R18C35 | PTF | BUH 2.2 |
| R3C36 | PTF | BUH 2.3 |
| R4C36 | PTF | HRH 0.0 |
| R4C36 | PTF | HRH 0.2 |
| R6C36 | PTF | BUH 2.3 |
| R27C36 | PTF | BUH 2.1 |
| R5C37 | PTF | BUH 2.4 |
| R9C37 | PTF | BUH 1.9 |
| R7C38 | PTF | BUH 2.2 |
| R25C38 | PTF | BUH 2.0 |
| R4C39 | PTF | BUH 2.4 |
| R13C39 | PTF | BUH 2.3 |
| R22C39 | PTF | BUH 2.4 |
| R8C40 | PTF | BUH 2.0 |
| R9C40 | PTF | BUH 2.3 |
| R11C40 | PTF | BUH 2.3 |
| R9C41 | PTF | BUH 2.3 |
| R14C41 | PTF | BUH 2.3 |
| R24C41 | PTF | BUH 1.9 |
| R33C41 | PTF | BUH 1.8 |
| R11C42 | PTF | BUH 2.5 |
| R12C42 | PTF | BUH 1.9 |
| R14C42 | PTF | BUH 2.3 |
| R20C42 | PTF | BUH 2.3 |
| R11C43 | PTF | BUH 2.0 |
| R22C43 | PTF | BUH 2.5 |
| R27C43 | PTF | BUH 1.9 |
| R6C44 | PTF | BUH 2.6 |
| R8C44 | PTF | BUH 2.3 |
| R9C44 | PTF | BUH 2.0 |
| R14C44 | PTF | BUH 2.4 |
| R27C44 | PTF | BUH 2.3 |
| R2C45 | PTF | BUH 2.2 |
| R15C45 | PTF | BUH 2.4 |
| R21C45 | PTF | BUH 2.1 |
| R4C46 | PTF | BUH 1.4 |
| R15C46 | PTF | BUH 2.1 |
| R22C46 | PTF | BUH 2.4 |
| R24C46 | PTF | BUH 2.3 |
| R27C46 | PTF | BUH 2.0 |
| R11C47 | PTF | BUH 2.0 |
| R27C47 | PTF | BUH 2.1 |
| R7C48 | PTF | BUH 2.0 |
| R16C48 | PTF | BUH 2.2 |
| R22C48 | PTF | BUH 2.2 |
| R24C48 | PTF | BUH 2.0 |
| R26C48 | PTF | BUH 2.1 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R29C48 | PTF | BUH 2.2 |
| R10C49 | PTF | BUH 2.3 |
| R13C49 | PTF | BUH 2.1 |
| R7C50 | PTF | BUH 2.2 |
| R8C50 | PTF | BUH 2.3 |
| R9C50 | PTF | BUH 2.4 |
| R10C50 | PTF | BUH 2.3 |
| R11C50 | PTF | BUH 2.2 |
| R13C50 | PTF | BUH 2.1 |
| R14C50 | PTF | BUH 2.2 |
| R19C50 | PTF | BUH 2.0 |
| R13C51 | PTF | BUH 2.2 |
| R16C51 | PTF | BUH 2.3 |
| R18C51 | PTF | BUH 2.5 |
| R24C51 | PTF | BUH 2.0 |
| R25C51 | PTF | BUH 2.2 |
| R8C52 | PTF | BUH 2.2 |
| R9C52 | PTF | BUH 2.0 |
| R16C52 | PTF | BUH 2.1 |
| R22C52 | PTF | BUH 2.3 |
| R25C52 | PTF | BUH 2.0 |
| R7C53 | PTF | BUH 2.2 |
| R13C53 | PTF | BUH 2.3 |
| R16C53 | PTF | BUH 2.2 |
| R17C53 | PTF | BUH 2.1 |
| R22C53 | PTF | BUH 2.0 |
| R25C53 | PTF | BUH 2.2 |
| R2C54 | PTF | BUH 2.0 |
| R5C54 | PTF | BUH 2.2 |
| R10C54 | PTF | BUH 1.9 |
| R12C54 | PTF | BUH 2.0 |
| R15C54 | PTF | BUH 2.2 |
| R16C54 | PTF | BUH 2.1 |
| R17C54 | PTF | BUH 0.7 |
| R18C54 | PTF | HRH 0.2 |
| R20C54 | PTF | BUH 2.2 |
| R21C54 | PTF | BUH 1.9 |
| R22C54 | PTF | BUH 2.1 |
| R28C54 | PTF | BUH 2.3 |
| R2C55 | PTF | BUH 2.0 |
| R6C55 | PTF | BUH 2.1 |
| R12C55 | PTF | BUH 2.1 |
| R7C56 | PTF | HRH 0.3 |
| R13C56 | PTF | BUH 2.3 |
| R6C57 | PTF | BUH 2.4 |
| R11C57 | PTF | BUH 2.0 |
| R12C57 | PTF | BUH 2.3 |
| R13C57 | PTF | BUH 2.0 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| **TUBE** | **DEFECT** | **LOCATION** |
| R15C57 | PTF | BUH 2.2 |
| R17C57 | PTF | BUH 2.2 |
| R27C57 | PTF | BUH 2.2 |
| R4C58 | PTF | BUH 2.6 |
| R5C58 | PTF | BUH 2.4 |
| R8C58 | PTF | BUH 2.2 |
| R17C58 | PTF | BUH 2.4 |
| R19C58 | PTF | BUH 2.2 |
| R30C58 | PTF | BUH 2.3 |
| R7C59 | PTF | BUH 2.5 |
| R8C59 | PTF | BUH 2.2 |
| R5C60 | PTF | BUH 2.5 |
| R19C60 | PTF | BUH 2.3 |
| R21C60 | PTF | BUH 2.4 |
| R7C61 | PTF | BUH 2.4 |
| R9C61 | PTF | BUH 2.4 |
| R10C61 | PTF | BUH 2.5 |
| R11C61 | PTF | BUH 2.4 |
| R18C61 | PTF | BUH 2.5 |
| R4C62 | PTF | BUH 2.4 |
| R7C62 | PTF | BUH 2.5 |
| R11C62 | PTF | BUH 2.5 |
| R7C63 | PTF | BUH 2.0 |
| R16C63 | PTF | BUH 2.2 |
| R18C63 | PTF | BUH 1.8 |
| R24C63 | PTF | BUH 1.9 |
| R7C64 | PTF | BUH 2.1 |
| R9C64 | PTF | HRH 0.2 |
| R12C64 | PTF | BUH 2.0 |
| R14C64 | PTF | BUH 2.0 |
| R17C64 | PTF | BUH 2.2 |
| R4C65 | PTF | BUH 2.1 |
| R7C65 | PTF | BUH 2.3 |
| R9C65 | PTF | BUH 2.2 |
| R8C66 | PTF | BUH 2.1 |
| R15C66 | PTF | BUH 2.4 |
| R12C67 | PTF | BUH 2.3 |
| R15C67 | PTF | BUH 2.1 |
| R15C68 | PTF | BUH 2.1 |
| R5C69 | PTF | BUH 2.2 |
| R7C69 | PTF | BUH 2.1 |
| R9C69 | PTF | BUH 2.1 |
| R11C69 | PTF | BUH 1.9 |
| R14C69 | PTF | BUH 2.4 |
| R15C69 | PTF | BUH 2.0 |
| R3C70 | PTF | BUH 2.1 |
| R4C70 | PTF | BUH 2.3 |
| R11C70 | PTF | BUH 2.1 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R13C74 | PTF | BUH 0.7 |
| R10C76 | PTF | BUH 2.0 |
| R9C77 | PTF | BUH 2.3 |
| R15C3 | SAI | TEH 17.2 |
| R16C5 | MAI | TEH 2.6 |
| R16C5 | MAI | TEH 6.5 |
| R16C5 | SCI | TEH 15.6 |
| R20C9 | MAI | TEH 3.6 |
| R20C9 | MAI | TEH 8.0 |
| R18C10 | MAI | TEH 4.2 |
| R18C10 | SAI | TEH 16.9 |
| R18C10 | MAI | TEH 18.1 |
| R7C13 | SAI | TEH 5.4 |
| R7C13 | MAI | TEH 17.2 |
| R7C13 | MAI | TEH 18.6 |
| R2C16 | SVI | TSH 0.0 |
| R3C16 | SVI | TSH 0.0 |
| R4C16 | SVI | TSH -0.1 |
| R5C16 | SVI | TSH -0.1 |
| R3C17 | SVI | TSH 0.0 |
| R11C17 | SVI | TSH -0.2 |
| R33C17 | MAI | TEH 7.3 |
| R33C17 | SAI | TEH 17.6 |
| R33C17 | SAI | TEH 17.8 |
| R1C18 | SAI | TSH 0.1 |
| R3C18 | SVI | TSH -0.2 |
| R3C19 | SVI | TSH -0.1 |
| R1C22 | SAI | TSH 0.1 |
| R1C24 | SAI | TSH 0.1 |
| R39C28 | MAI | TEH 3.0 |
| R39C28 | SAI | TEH 17.8 |
| R40C40 | MAI | TEH 2.7 |
| R43C44 | MAI | TEH 2.6 |
| R43C44 | SAI | TEH 16.5 |
| R43C44 | MAI | 01H 0.1 |
| R45C52 | MAI | TEH 2.5 |
| R45C52 | SAI | TSH 0.2 |
| R45C52 | SAI | TSH 0.2 |
| R42C53 | MAI | TEH 2.7 |
| R34C60 | MAI | TEH 3.0 |
| R43C60 | SAI | TEH 3.1 |
| R43C60 | SAI | TEH 16.7 |
| R43C60 | SAI | TEH 17.6 |
| R31C65 | SAI | TEH 4.7 |
| R31C65 | SAI | TEH 17.0 |
| R31C65 | SAI | TEH 17.6 |
| R39C67 | MAI | TEH 3.3 |
| R39C67 | SAI | TEH 16.9 |

| TUBES PLUGGED IN STEAM GENERATOR A | | |
|---|---|---|
| **TUBE** | **DEFECT** | **LOCATION** |
| R39C67 | SAI | TEH 17.8 |
| R5C71 | SAI | TEH [illegible].2 |
| R3C72 | SAI | TSH 0.2 |
| R3C72 | SAI | TSH 0.3 |
| R3C74 | SVI | TSH 0.2 |
| R6C75 | MAI | TEH 8.6 |
| R18C75 | SAI | TEH 6.6 |
| R18C75 | SAI | TEH 15.8 |
| R3C76 | SVI | TSH 0.1 |
| R4C76 | SVI | TSH 0.1 |
| R31C76 | SAI | TEH 2.6 |
| R31C76 | MAI | TEH 2.6 |
| R33C76 | SAI | TEH 3.2 |
| R33C76 | MAI | TEH 16.7 |
| R33C76 | SAI | TEH 18.5 |
| R3C77 | SVI | TSH 0.4 |
| R4C77 | MVI | TSH 0.2 |
| R10C79 | SAI | TEH 11.7 |
| R21C80 | MAI | TEH 2.4 |
| R21C80 | SAI | TEH 4.3 |
| R21C80 | SAI | TEH 17.5 |
| R12C84 | MAI | TEH 2.7 |
| R12C84 | SAI | TEH 15.3 |
| R12C84 | SAI | TSH 0.2 |
| R12C84 | SAI | TSH 0.3 |
| R13C85 | MAI | TEH 6.0 |
| R13C85 | SAI | TEH 18.7 |
| R18C5 | 48 | 01H -0.1 |
| R22C7 | 41 | 02H 0.1 |
| R27C11 | 41 | 01H -0.1 |
| R33C16 | 71 | 01C -0.0 |
| R29C81 | 42 | 05H 0.2 |

| TUBES PLUGGED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R23C29 | 41 | 01C -0.03 |
| R28C51 | 40 | 01C -0.03 |
| R8C2 | MAI | TEH 5.5 |
| R8C2 | MAI | TEH 16.8 |
| R12C3 | SAI | 01H 0.8 |
| R1C4 | SAI | TEH 8.3 |
| R9C6 | MAI | TEH 7.6 |
| R8C7 | MAI | TEH 8.7 |
| R1C9 | SAI | TEH 8.1 |
| R1C10 | MAI | TEH 2.7 |
| R23C13 | MAI | TEH 2.4 |
| R23C13 | MAI | TEH 17.0 |

| TUBES PLUGGED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R4C19 | SAI | TSH 0.2 |
| R1C27 | MAI | TEH 2.8 |
| R1C27 | MAI | TEH 16.6 |
| R1C28 | MAI | TEH 2.3 |
| R1C28 | SAI | TEH 17.8 |
| R1C31 | MAI | TEH 2.8 |
| R1C31 | SAI | TSH 0.4 |
| R1C41 | SAI | TEH 3.5 |
| R1C41 | MAI | TEH 17.4 |
| R34C57 | MAI | TEH 5.8 |
| R33C60 | MAI | TEH 6.1 |
| R33C60 | SAI | TEH 17.1 |
| R36C60 | MAI | TEH 5.7 |
| R33C61 | MAI | TEH 5.0 |
| R33C61 | MAI | TEH 16.9 |
| R33C62 | MAI | TEH 6.0 |
| R33C64 | MAI | TEH 4.0 |
| R36C65 | SAI | TEH 4.4 |
| R36C65 | SAI | TEH 9.8 |
| R36C65 | SAI | TEH 16.8 |
| R33C69 | MAI | TEH 5.8 |
| R33C69 | SAI | TEH 17.0 |
| R33C70 | MAI | TEH 5.5 |
| R1C71 | MAI | TEH 4.0 |
| R1C71 | MAI | TEH 17.0 |
| R33C72 | MAI | TEH 5.4 |
| R33C73 | MAI | TEH 6.4 |
| R33C73 | SAI | TEH 17.1 |
| R1C75 | SAI | TEH 4.6 |
| R1C75 | SAI | TEH 17.0 |
| R5C75 | MAI | TEH 2.6 |
| R5C75 | SAI | TEH 17.3 |
| R33C75 | MAI | TEH 3.0 |
| R33C75 | MAI | TEH 5.4 |
| R33C75 | SAI | TEH 16.7 |
| R9C76 | MAI | TEH 3.9 |
| R9C76 | SAI | TEH 15.9 |
| R5C77 | MAI | TEH 3.0 |
| R5C77 | SAI | TEH 17.7 |
| R6C77 | MAI | TEH 6.4 |
| R6C77 | MAI | TEH 14.5 |
| R16C78 | MAI | TEH 2.3 |
| R16C78 | SAI | TEH 16.9 |
| R29C78 | MAI | TEH 6.6 |
| R29C78 | SAI | TEH 17.2 |
| R17C79 | MAI | TEH 3.3 |
| R17C79 | SAI | TEH 16.3 |
| R22C84 | MAI | TEH 3.0 |

| TUBES PLUGGED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R22C84 | MAI | TEH 17.5 |
| R24C84 | MAI | TEH 3.0 |
| R24C84 | MAI | TEH 17.9 |
| R23C85 | MAI | TEH 4.2 |
| R23C85 | SAI | TEH 17.9 |
| R1C89 | MAI | TEH 2.4 |
| R1C89 | SAI | TEH 17.1 |
| R11C90 | SAI | 01H 1.8 |
| R18C17 | PTF | BUH 2.2 |
| R5C20 | PTF | BUH 2.3 |
| R18C22 | PTF | BUH 2.6 |
| R9C27 | PTF | BUH 2.1 |
| R14C29 | PTF | BUH 2.1 |
| R11C30 | PTF | BUH 2.1 |
| R11C30 | PTF | BUH 2.3 |
| R14C30 | PTF | BUH 2.4 |
| R16C30 | PTF | BUH 2.5 |
| R21C30 | PTF | BUH 2.4 |
| R8C31 | PTF | BUH 2.5 |
| R12C31 | PTF | BUH 2.2 |
| R16C31 | PTF | BUH 2.6 |
| R8C32 | PTF | BUH 2.3 |
| R11C32 | PTF | BUH 1.8 |
| R12C32 | PTF | BUH 2.4 |
| R13C32 | PTF | BUH 1.9 |
| R16C32 | PTF | BUH 2.5 |
| R2C33 | PTF | BUH 2.9 |
| R8C33 | PTF | BUH 2.2 |
| R18C33 | PTF | BUH 2.3 |
| R12C34 | PTF | BUH 2.4 |
| R18C34 | PTF | BUH 2.2 |
| R18C34 | PTF | BUH 3.0 |
| R14C35 | PTF | BUH 2.5 |
| R16C35 | PTF | BUH 2.4 |
| R16C36 | PTF | BUH 2.4 |
| R21C36 | PTF | BUH 2.4 |
| R14C37 | PTF | BUH 2.5 |
| R16C37 | PTF | BUH 2.5 |
| R20C37 | PTF | BUH 2.5 |
| R8C38 | PTF | BUH 2.5 |
| R14C38 | PTF | BUH 2.1 |
| R21C38 | PTF | BUH 2.4 |
| R13C39 | PTF | BUH 2.4 |
| R20C39 | PTF | BUH 2.4 |
| R21C39 | PTF | BUH 2.0 |
| R21C39 | PTF | BUH 2.1 |
| R21C39 | PTF | BUH 3.5 |
| R23C39 | PTF | BUH 2.1 |

| TUBES PLUGGED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R23C39 | PTF | BUH 2.2 |
| R14C40 | PTF | BUH 2.3 |
| R21C40 | PTF | BUH 2.5 |
| R16C41 | PTF | BUH 2.5 |
| R18C42 | PTF | BUH 2.5 |
| R23C42 | PTF | BUH 2.1 |
| R18C44 | PTF | BUH 2.1 |
| R21C44 | PTF | BUH 2.5 |
| R12C45 | PTF | BUH 2.4 |
| R12C45 | PTF | BUH 2.5 |
| R13C45 | PTF | BUH 2.5 |
| R16C45 | PTF | BUH 2.5 |
| R16C46 | PTF | BUH 2.1 |
| R20C47 | PTF | BUH 2.5 |
| R23C48 | PTF | BUH 1.9 |
| R24C48 | PTF | BUH 2.8 |
| R23C49 | PTF | BUH 2.1 |
| R23C50 | PTF | BUH 2.2 |
| R9C51 | PTF | BUH 2.2 |
| R24C51 | PTF | BUH 2.9 |
| R13C53 | PTF | BUH 2.6 |
| R9C54 | PTF | BUH 2.5 |
| R24C54 | PTF | BUH 2.2 |
| R4C56 | PTF | BUH 2.1 |
| R11C56 | PTF | BUH 2.2 |
| R13C56 | PTF | BUH 2.2 |
| R18C57 | PTF | BUH 2.3 |
| R11C59 | PTF | BUH 2.4 |
| R17C61 | PTF | BUH 2.6 |
| R13C63 | PTF | BUH 2.1 |
| R13C65 | PTF | BUH 2.7 |
| R16C66 | PTF | BUH 2.5 |
| R16C66 | PTF | BUH 2.6 |
| R18C66 | PTF | BUH 2.5 |
| R9C75 | SVI | TSH 0.4 |
| R3C79 | SVI | TS 0.3 |
| R22C86 | SVI | 01C -0.0 |
| R14C89 | SVI | 03C 0.1 |
| R13C90 | SVI | TSH 1.3 |
| R26C20 | ND | NA |
| R30C20 | ND | NA |
| R32C20 | ND | NA |
| R30C23 | ND | NA |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R4C2 | SAI | TEH 2.6 |
| R4C2 | SAI | TEH 2.7 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R4C2 | SAI | TEH 3.3 |
| R7C2 | SAI | TEH 6.6 |
| R7C2 | SAI | TEH 6.8 |
| R4C3 | MAI | TEH 3.1 |
| R4C3 | MAI | TEH 3.3 |
| R7C3 | MAI | TEH 4.8 |
| R7C3 | MAI | TEH 12.1 |
| R9C3 | SAI | TEH 3.5 |
| R9C3 | SAI | TEH 5.5 |
| R9C3 | SAI | TEH 16.2 |
| R9C4 | MAI | TEH 4.4 |
| R9C4 | MAI | TEH 4.5 |
| R6C5 | MAI | TEH 3.0 |
| R6C5 | SAI | TEH 16.0 |
| R8C5 | MAI | TEH 2.5 |
| R8C5 | SAI | TEH 10.8 |
| R8C5 | SAI | TEH 16.3 |
| R8C5 | SAI | TEH 16.4 |
| R9C5 | MAI | TEH 2.7 |
| R9C5 | SAI | TEH 7.8 |
| R9C5 | SAI | TEH 11.0 |
| R9C5 | SAI | TEH 16.3 |
| R11C5 | SAI | TEH 6.7 |
| R11C5 | SAI | TEH 9.6 |
| R11C5 | SAI | TEH 16.1 |
| R11C5 | SAI | TEH 16.1 |
| R13C5 | SAI | TEH 11.3 |
| R13C5 | SAI | TEH 16.0 |
| R3C6 | SAI | TEH 2.6 |
| R3C6 | SAI | TEH 2.8 |
| R9C6 | MAI | TEH 3.6 |
| R9C6 | SAI | TEH 16.0 |
| R10C6 | MAI | TEH 2.7 |
| R10C6 | SAI | TEH 11.1 |
| R10C6 | MAI | TEH 11.3 |
| R12C6 | SAI | TEH 3.6 |
| R12C6 | SAI | TEH 3.7 |
| R14C6 | SAI | TEH 5.5 |
| R14C6 | SAI | TEH 6.0 |
| R16C6 | MAI | TEH 8.5 |
| R16C6 | MAI | TEH 8.7 |
| R17C6 | SAI | TEH 6.2 |
| R17C6 | SAI | TEH 16.0 |
| R17C6 | SAI | TEH 16.8 |
| R19C6 | SAI | TEH 6.1 |
| R19C6 | SAI | TEH 6.4 |
| R6C7 | MAI | TEH 2.6 |
| R6C7 | MAI | TEH 2.8 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R8C7 | SAI | TEH 8.4 |
| R8C7 | SAI | TEH 12.1 |
| R10C7 | MAI | TEH 1.8 |
| R10C7 | SAI | TEH 16.6 |
| R14C7 | MAI | TEH 3.2 |
| R14C7 | SAI | TEH 7.5 |
| R14C7 | SAI | TEH 9.2 |
| R14C7 | MAI | TEH 10.5 |
| R19C7 | MAI | TEH 4.3 |
| R19C7 | MAI | TEH 13.1 |
| R9C8 | MAI | TEH 4.2 |
| R9C8 | SAI | TEH 15.9 |
| R12C8 | SAI | TEH 4.9 |
| R12C8 | SAI | TEH 5.0 |
| R13C8 | MAI | TEH 3.3 |
| R13C8 | SAI | TEH 16.0 |
| R17C8 | MAI | TEH 2.4 |
| R17C8 | MAI | TEH 2.9 |
| R19C8 | SAI | TEH 8.0 |
| R19C8 | SAI | TEH 9.1 |
| R2C9 | MAI | TEH 3.2 |
| R2C9 | MAI | TEH 15.9 |
| R6C9 | MAI | TEH 3.8 |
| R6C9 | MAI | TEH 15.9 |
| R11C9 | SAI | TEH 2.9 |
| R11C9 | SAI | TEH 6.0 |
| R17C9 | SAI | TEH 5.4 |
| R17C9 | SAI | TEH 8.6 |
| R3C10 | MAI | TEH 2.7 |
| R3C10 | MAI | TEH 2.7 |
| R6C10 | SAI | TEH 8.0 |
| R6C10 | SAI | TEH 11.8 |
| R6C10 | SAI | TEH 16.0 |
| R8C10 | SAI | TEH 7.3 |
| R8C10 | SAI | TEH 8.1 |
| R8C10 | SAI | TEH 8.3 |
| R14C10 | SAI | TEH 8.3 |
| R14C10 | MAI | TEH 9.6 |
| R14C10 | MAI | TEH 10.2 |
| R16C10 | SAI | TEH 8.4 |
| R16C10 | SAI | TEH 9.0 |
| R21C10 | SAI | TEH 5.7 |
| R21C10 | SAI | TEH 6.1 |
| R27C10 | MAI | TEH 2.7 |
| R27C10 | MAI | TEH 2.9 |
| R14C11 | SAI | TEH 2.9 |
| R14C11 | SAI | TEH 4.4 |
| R19C11 | SAI | TEH 6.8 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R19C11 | SAI | TEH 8.0 |
| R20C11 | SAI | TEH 7.6 |
| R20C11 | SAI | TEH 9.7 |
| R21C11 | MAI | TEH 4.1 |
| R21C11 | MAI | TEH 5.5 |
| R11C12 | SAI | TEH 5.3 |
| R11C12 | SAI | TEH 6.7 |
| R11C12 | SAI | TEH 17.1 |
| R11C12 | SAI | TEH 17.1 |
| R20C12 | MAI | TEH 4.0 |
| R20C12 | SAI | TEH 16.1 |
| R4C13 | SAI | TEH 2.5 |
| R4C13 | SAI | TEH 2.6 |
| R10C13 | SAI | TEH 8.3 |
| R10C13 | SAI | TEH 16.3 |
| R12C13 | MAI | TEH 4.4 |
| R12C13 | SAI | TEH 10.1 |
| R12C13 | MAI | TEH 16.1 |
| R17C13 | SAI | TEH 6.0 |
| R17C13 | SAI | TEH 6.6 |
| R17C13 | MAI | TEH 7.6 |
| R17C13 | SAI | TEH 8.3 |
| R18C13 | MAI | TEH 4.3 |
| R18C13 | MAI | TEH 5.6 |
| R3C14 | MAI | TEH 6.9 |
| R3C14 | SAI | TEH 16.1 |
| R8C14 | MAI | TEH 5.1 |
| R8C14 | MAI | TEH 5.8 |
| R18C14 | MAI | TEH 3.5 |
| R18C14 | MAI | TEH 16.1 |
| R18C15 | MAI | TEH 7.6 |
| R18C15 | MAI | TEH 9.6 |
| R21C15 | MAI | TEH 3.7 |
| R21C15 | MAI | TEH 16.0 |
| R17C16 | SAI | TEH 7.0 |
| R17C16 | SAI | TEH 7.3 |
| R21C17 | MAI | TEH 9.2 |
| R21C17 | SAI | TEH 15.9 |
| R26C18 | SAI | TEH 8.3 |
| R26C18 | SAI | TEH 8.6 |
| R27C18 | SAI | TEH 4.3 |
| R27C18 | SAI | TEH 5.6 |
| R27C18 | SAI | TEH 5.8 |
| R23C19 | SAI | TEH 4.8 |
| R23C19 | SAI | TEH 5.0 |
| R34C19 | SAI | TEH 4.8 |
| R34C19 | SAI | TEH 17.1 |
| R34C19 | SAI | TEH 17.2 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R35C20 | SAI | TEH 5.8 |
| R35C20 | SAI | TEH 15.8 |
| R31C21 | SAI | TEH 9.2 |
| R31C21 | SAI | TEH 16.1 |
| R33C21 | MAI | TEH 2.5 |
| R33C21 | MAI | TEH 2.9 |
| R37C21 | SAI | TEH 3.3 |
| R37C21 | SAI | TEH 4.0 |
| R29C22 | MAI | TEH 2.9 |
| R29C22 | MAI | TEH 3.0 |
| R30C22 | SAI | TEH 13.6 |
| R30C22 | SAI | TEH 13.6 |
| R32C22 | SAI | TEH 2.8 |
| R32C22 | SAI | TEH 2.9 |
| R31C23 | MAI | TEH 1.3 |
| R31C23 | MAI | TEH 2.7 |
| R31C23 | MAI | TEH 2.8 |
| R32C23 | MAI | TEH 2.8 |
| R36C23 | SAI | TEH 8.0 |
| R36C23 | SAI | TEH 8.9 |
| R35C24 | SAI | TEH 4.9 |
| R35C24 | SAI | TEH 6.9 |
| R36C24 | MAI | TEH 2.8 |
| R36C24 | MAI | TEH 8.1 |
| R28C25 | MAI | TEH 2.5 |
| R28C25 | SAI | TEH 2.9 |
| R29C25 | SAI | TEH 3.6 |
| R29C25 | SAI | TEH 4.0 |
| R35C25 | MAI | TEH 2.4 |
| R35C25 | MAI | TEH 2.5 |
| R1C26 | SAI | TEH 4.6 |
| R1C26 | SAI | TEH 5.0 |
| R1C26 | MAI | TEH 11.5 |
| R30C26 | SAI | TEH 10.5 |
| R30C26 | SAI | TEH 11.2 |
| R32C26 | MAI | TEH 2.4 |
| R32C26 | MAI | TEH 2.7 |
| R39C26 | MAI | TEH 2.5 |
| R39C26 | SAI | TEH 3.7 |
| R39C26 | SAI | TEH 4.7 |
| R1C27 | MAI | TEH 2.3 |
| R1C27 | MAI | TEH 16.0 |
| R34C27 | MAI | TEH 2.6 |
| R34C27 | MAI | TEH 4.8 |
| R34C27 | SAI | TEH 9.9 |
| R34C27 | MAI | TEH 15.8 |
| R31C28 | MAI | TEH 2.4 |
| R31C28 | MAI | TEH 2.8 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R32C29 | MAI | TEH 2.6 |
| R32C29 | MAI | TEH 2.8 |
| R34C29 | SAI | TEH 4.5 |
| R34C29 | SAI | TEH 5.2 |
| R36C29 | SAI | TEH 2.9 |
| R36C29 | SAI | TEH 3.5 |
| R37C29 | SAI | TEH 3.8 |
| R37C29 | SAI | TEH 4.5 |
| R39C29 | MAI | TEH 3.5 |
| R39C29 | MAI | TEH 8.4 |
| R39C30 | MAI | TEH 4.0 |
| R39C30 | MAI | TEH 8.4 |
| R34C31 | SAI | TEH 4.1 |
| R34C31 | SAI | TEH 4.4 |
| R35C31 | MAI | TEH 2.7 |
| R35C31 | MAI | TEH 15.6 |
| R36C31 | MAI | TEH 2.5 |
| R36C31 | MAI | TEH 2.7 |
| R37C31 | MAI | TEH 2.6 |
| R37C31 | MAI | TEH 3.0 |
| R38C31 | MAI | TEH 4.6 |
| R38C31 | MAI | TEH 6.8 |
| R32C32 | MAI | TEH 2.6 |
| R32C32 | MAI | TEH 2.9 |
| R33C32 | MAI | TEH 2.5 |
| R33C32 | MAI | TEH 2.7 |
| R40C32 | MAI | TEH 3.2 |
| R40C32 | MAI | TEH 4.1 |
| R40C33 | MAI | TEH 2.8 |
| R40C33 | MAI | TEH 3.6 |
| R36C34 | SAI | TEH 3.2 |
| R36C34 | SAI | TEH 3.3 |
| R39C35 | SAI | TEH 3.6 |
| R39C35 | SAI | TEH 6.5 |
| R38C36 | MAI | TEH 2.9 |
| R38C36 | SAI | TEH 8.7 |
| R39C36 | SAI | TEH 3.5 |
| R39C36 | SAI | TEH 6.7 |
| R37C37 | SAI | TEH 2.9 |
| R37C37 | SAI | TEH 3.0 |
| R37C38 | MAI | TEH 1.6 |
| R37C38 | MAI | TEH 7.3 |
| R39C38 | SAI | TEH 1.4 |
| R39C38 | SAI | TEH 1.4 |
| R43C38 | MAI | TEH 1.2 |
| R43C38 | MAI | TEH 1.4 |
| R34C39 | SAI | TEH 2.8 |
| R34C39 | MAI | TEH 3.1 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R34C39 | SAI | TEH 4.1 |
| R39C39 | MAI | TEH 3.2 |
| R39C39 | MAI | TEH 8.5 |
| R41C39 | MAI | TEH 2.9 |
| R41C39 | SAI | TEH 15.6 |
| R38C40 | MAI | TEH 2.9 |
| R38C40 | SAI | TEH 4.2 |
| R42C40 | SAI | TEH 7.9 |
| R42C40 | SAI | TEH 8.6 |
| R35C41 | MAI | TEH 2.8 |
| R35C41 | MAI | TEH 3.0 |
| R37C41 | SAI | TEH 4.3 |
| R37C41 | SAI | TEH 4.5 |
| R38C41 | SAI | TEH 6.6 |
| R38C41 | SAI | TEH 6.7 |
| R40C41 | SAI | TEH 4.7 |
| R40C41 | SAI | TEH 8.5 |
| R41C41 | SAI | TEH 7.3 |
| R41C41 | SAI | TEH 8.6 |
| R42C41 | SAI | TEH 2.7 |
| R42C41 | SAI | TEH 2.9 |
| R35C42 | MAI | TEH 2.9 |
| R35C42 | MAI | TEH 7.2 |
| R39C42 | MAI | TEH 5.7 |
| R39C42 | MAI | TEH 6.9 |
| R34C43 | MAI | TEH 1.8 |
| R34C43 | MAI | TEH 2.7 |
| R35C43 | SAI | TEH 2.5 |
| R35C43 | SAI | TEH 6.4 |
| R35C43 | SAI | TEH 6.6 |
| R37C43 | MAI | TEH 7.4 |
| R37C43 | MAI | TEH 8.4 |
| R39C43 | SAI | TEH 5.4 |
| R39C43 | SAI | TEH 10.3 |
| R40C43 | MAI | TEH 2.9 |
| R40C43 | SAI | TEH 12.1 |
| R41C43 | MAI | TEH 2.8 |
| R41C43 | MAI | TEH 6.3 |
| R43C43 | SAI | TEH 3.3 |
| R43C43 | SAI | TEH 9.3 |
| R43C43 | SAI | TEH 10.3 |
| R44C43 | SAI | TEH 2.6 |
| R44C43 | SAI | TEH 2.7 |
| R35C44 | SAI | TEH 2.4 |
| R35C44 | SAI | TEH 2.7 |
| R36C44 | SAI | TEH 2.4 |
| R36C44 | SAI | TEH 2.6 |
| R44C44 | MAI | TEH 2.7 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R44C44 | MAI | TEH 9.8 |
| R34C45 | SAI | TEH 5.5 |
| R34C45 | SAI | TEH 6.1 |
| R35C45 | SAI | TEH 4.1 |
| R35C45 | SAI | TEH 5.7 |
| R36C45 | SAI | TEH 4.9 |
| R36C45 | SAI | TEH 5.4 |
| R37C45 | MAI | TEH 2.3 |
| R37C45 | MAI | TEH 2.6 |
| R39C45 | SAI | TEH 2.7 |
| R39C45 | SAI | TEH 2.8 |
| R44C45 | SAI | TEH 7.8 |
| R44C45 | SAI | TEH 8.4 |
| R37C46 | SAI | TEH 1.3 |
| R37C46 | SAI | TEH 2.7 |
| R40C46 | MAI | TEH 2.9 |
| R40C46 | SAI | TEH 16.1 |
| R41C46 | MAI | TEH 2.8 |
| R41C46 | MAI | TEH 5.5 |
| R43C46 | MAI | TEH 4.6 |
| R43C46 | MAI | TEH 9.5 |
| R44C46 | SAI | TEH 3.1 |
| R44C46 | SAI | TEH 16.6 |
| R44C46 | SAI | TEH 16.8 |
| R35C47 | MAI | TEH 2.7 |
| R35C47 | MAI | TEH 2.8 |
| R36C47 | MAI | TEH 2.3 |
| R36C47 | MAI | TEH 2.7 |
| R38C47 | SAI | TEH 2.8 |
| R38C47 | SAI | TEH 4.1 |
| R38C47 | SAI | TEH 15.8 |
| R41C47 | MAI | TEH 1.3 |
| R41C47 | MAI | TEH 2.6 |
| R41C47 | MAI | TEH 6.7 |
| R43C47 | MAI | TEH 1.2 |
| R43C47 | SAI | TEH 9.1 |
| R38C48 | MAI | TEH 2.4 |
| R38C48 | MAI | TEH 2.6 |
| R40C48 | MAI | TEH 4.4 |
| R40C48 | MAI | TEH 6.7 |
| R41C48 | SAI | TEH 2.6 |
| R41C48 | SAI | TEH 2.7 |
| R34C49 | SAI | TEH 4.5 |
| R34C49 | SAI | TEH 16.0 |
| R37C49 | MAI | TEH 3.2 |
| R37C49 | MAI | TEH 7.7 |
| R40C49 | MAI | TEH 2.9 |
| R40C49 | SAI | TEH 17.0 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R40C49 | MAI | TEH 17.1 |
| R45C49 | SAI | TEH 2.3 |
| R45C49 | SAI | TEH 2.8 |
| R33C50 | MAI | TEH 2.3 |
| R33C50 | MAI | TEH 2.6 |
| R35C50 | SAI | TEH 4.9 |
| R35C50 | SAI | TEH 5.4 |
| R37C50 | SAI | TEH 3.7 |
| R37C50 | SAI | TEH 4.8 |
| R38C50 | MAI | TEH 2.5 |
| R38C50 | SAI | TEH 2.9 |
| R38C50 | SAI | TEH 15.9 |
| R39C50 | SAI | TEH 3.1 |
| R39C50 | SAI | TEH 8.7 |
| R39C50 | SAI | TEH 8.9 |
| R43C50 | MAI | TEH 6.1 |
| R43C50 | MAI | TEH 7.2 |
| R34C51 | MAI | TEH 2.6 |
| R34C51 | MAI | TEH 2.6 |
| R34C51 | SAI | TEH 13.1 |
| R34C51 | SAI | TEH 13.1 |
| R34C51 | SAI | TEH !3.4 |
| R36C51 | SAI | TEH 4.6 |
| R36C51 | SAI | TEH 5.3 |
| R37C51 | SAI | TEH 5.2 |
| R37C51 | SAI | TEH 11.3 |
| R39C51 | MAI | TEH 3.3 |
| R39C51 | SAI | TEH 7.6 |
| R39C51 | SAI | TEH 7.9 |
| R40C51 | SAI | TEH 5.7 |
| R40C51 | SAI | TEH 5.8 |
| R42C51 | SAI | TEH 2.4 |
| R42C51 | SAI | TEH 2.5 |
| R33C52 | SAI | TEH 3.0 |
| R33C52 | SAI | TEH 17.1 |
| R36C52 | SAI | TEH 9.3 |
| R36C52 | SAI | TEH 11.3 |
| R36C52 | SAI | TEH 11.6 |
| R44C52 | SAI | TEH 4.2 |
| R44C52 | SAI | TEH 4.4 |
| R37C53 | MAI | TEH 2.9 |
| R37C53 | MAI | TEH 16.0 |
| R40C53 | MAI | TEH 5.2 |
| R40C53 | MAI | TEH 5.3 |
| R44C53 | SAI | TEH 5.3 |
| R44C53 | SAI | TEH 7.9 |
| R35C54 | MAI | TEH 4.3 |
| R35C54 | MAI | TEH 11.1 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R37C54 | SAI | TEH 4.3 |
| R37C54 | SAI | TEH 4.7 |
| R40C54 | MAI | TEH 3.4 |
| R40C54 | MAI | TEH 4.8 |
| R34C55 | SAI | TEH 4.1 |
| R34C55 | SAI | TEH 16.2 |
| R37C55 | MAI | TEH 2.6 |
| R37C55 | MAI | TEH 2.8 |
| R38C55 | MAI | TEH 2.8 |
| R38C55 | SAI | TEH 9.7 |
| R39C55 | MAI | TEH 2.6 |
| R39C55 | MAI | TEH 5.1 |
| R39C55 | SAI | TEH 7.8 |
| R42C55 | SAI | TEH 3.2 |
| R42C55 | SAI | TEH 3.3 |
| R32C56 | MAI | TEH 2.7 |
| R32C56 | MAI | TEH 2.8 |
| R33C56 | SAI | TEH 4.2 |
| R33C56 | SAI | TEH 4.4 |
| R33C56 | SAI | TEH 8.5 |
| R37C57 | MAI | TEH 1.9 |
| R37C57 | SAI | TEH 16.8 |
| R37C57 | SAI | TEH 17.0 |
| R39C57 | MAI | TEH 2.6 |
| R39C57 | SAI | TEH 3.0 |
| R39C57 | SAI | TEH 5.2 |
| R40C57 | MAI | TEH 4.8 |
| R40C57 | SAI | TEH 16.2 |
| R41C57 | MAI | TEH 2.5 |
| R41C57 | MAI | TEH 5.2 |
| R41C57 | SAI | TEH 16.1 |
| R35C58 | SAI | TEH 5.5 |
| R35C58 | SAI | TEH 8.4 |
| R37C58 | MAI | TEH 5.6 |
| R37C58 | MAI | TEH 16.0 |
| R38C58 | MAI | TEH 3.3 |
| R38C58 | SAI | TEH 7.0 |
| R39C58 | SAI | TEH 4.6 |
| R39C58 | SAI | TEH 5.6 |
| R41C58 | MAI | TEH 2.7 |
| R41C58 | MAI | TEH 3.0 |
| R41C58 | SAI | TEH 16.8 |
| R41C58 | SAI | TEH 17.1 |
| R42C58 | MAI | TEH 5.7 |
| R42C58 | SAI | TEH 8.5 |
| R37C59 | MAI | TEH 3.0 |
| R37C59 | SAI | TEH 16.7 |
| R37C59 | SAI | TEH 16.9 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R38C60 | SAI | TEH 4.9 |
| R38C60 | SAI | TEH 5.1 |
| R31C61 | MAI | TEH 2.5 |
| R31C61 | SAI | TEH 16.0 |
| R34C61 | MAI | TEH 2.7 |
| R34C61 | MAI | TEH 2.7 |
| R39C61 | SAI | TEH 2.7 |
| R39C61 | MAI | TEH 2.8 |
| R41C61 | MAI | TEH 5.4 |
| R41C61 | SAI | TEH 16.7 |
| R41C61 | MAI | TEH 16.9 |
| R37C62 | MAI | TEH 3.0 |
| R37C62 | SAI | TEH 7.4 |
| R31C63 | SAI | TEH 3.9 |
| R31C63 | SAI | TEH 8.6 |
| R31C63 | SAI | TEH 9.8 |
| R39C63 | MAI | TEH 2.6 |
| R39C63 | SAI | TEH 16.6 |
| R39C63 | SAI | TEH 16.8 |
| R29C64 | SAI | TEH 3.4 |
| R29C64 | SAI | TEH 6.0 |
| R38C64 | MAI | TEH 2.8 |
| R38C64 | MAI | TEH 5.5 |
| R38C64 | SAI | TEH 15.8 |
| R29C65 | SAI | TEH 5.1 |
| R29C65 | SAI | TEH 5.4 |
| R30C65 | MAI | TEH 2.6 |
| R30C65 | SAI | TEH 16.7 |
| R30C65 | SAI | TEH 16.8 |
| R34C65 | MAI | TEH 2.8 |
| R34C65 | SAI | TEH 9.8 |
| R30C66 | SAI | TEH 3.9 |
| R30C66 | SAI | TEH 16.7 |
| R30C66 | SAI | TEH 17.1 |
| R32C66 | MAI | TEH 3.4 |
| R32C66 | MAI | TEH 16.6 |
| R32C66 | SAI | TEH 16.7 |
| R30C67 | SAI | TEH 4.9 |
| R30C67 | SAI | TEH 8.6 |
| R31C67 | SAI | TEH 3.3 |
| R31C67 | SAI | TEH 16.7 |
| R31C67 | SAI | TEH 17.1 |
| R32C67 | MAI | TEH 2.7 |
| R32C67 | MAI | TEH 2.9 |
| R35C67 | SAI | TEH 3.6 |
| R35C67 | SAI | TEH 16.3 |
| R35C67 | SAI | TEH 16.9 |
| R33C68 | SAI | TEH 6.7 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R33C68 | SAI | TEH 8.1 |
| R35C68 | MAI | TEH 2.4 |
| R35C68 | SAI | TEH 16.5 |
| R35C68 | MAI | TEH 16.8 |
| R37C68 | MAI | TEH 2.6 |
| R37C68 | SAI | TEH 8.1 |
| R37C68 | SAI | TEH 8.1 |
| R30C69 | SAI | TEH 6.5 |
| R30C69 | SAI | TEH 9.1 |
| R33C69 | MAI | TEH 2.6 |
| R33C69 | SAI | TEH 16.6 |
| R33C69 | SAI | TEH 16.7 |
| R36C69 | MAI | TEH 2.7 |
| R36C69 | MAI | TEH 2.7 |
| R38C69 | MAI | TEH 2.6 |
| R38C69 | MAI | TEH 5.9 |
| R38C69 | SAI | TEH 9.3 |
| R39C69 | SAI | TEH 4.4 |
| R39C69 | SAI | TEH 15.9 |
| R31C70 | MAI | TEH 3.6 |
| R31C70 | SAI | TEH 15.8 |
| R32C70 | SAI | TEH 6.7 |
| R32C70 | SAI | TEH 16.1 |
| R36C70 | MAI | TEH 2.4 |
| R36C70 | MAI | TEH 3.0 |
| R36C70 | SAI | TEH 10.8 |
| R39C70 | MAI | TEH 2.7 |
| R39C70 | MAI | TEH 2.7 |
| R30C71 | SAI | TEH 4.9 |
| R30C71 | SAI | TEH 6.8 |
| R30C71 | SAI | TEH 6.9 |
| R31C71 | MAI | TEH 2.8 |
| R31C71 | SAI | TEH 15.9 |
| R33C71 | MAI | TEH 4.6 |
| R33C71 | SAI | TEH 8.8 |
| R34C71 | SCI | TEH 2.2 |
| R34C71 | SCI | TEH 2.3 |
| R34C71 | MAI | TEH 3.1 |
| R36C71 | SAI | TEH 3.1 |
| R36C71 | SAI | TEH 3.2 |
| R38C71 | SAI | TEH 3.5 |
| R38C71 | SAI | TEH 3.8 |
| R6C72 | SAI | TEH 6.8 |
| R6C72 | SAI | TEH 8.0 |
| R29C72 | SCI | TEH 2.2 |
| R29C72 | SCI | TEH 2.7 |
| R32C72 | SAI | TEH 2.6 |
| R32C72 | SAI | TEH 2.9 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R35C72 | SAI | TEH 4.1 |
| R35C72 | SAI | TEH 11.5 |
| R28C73 | MAI | TEH 2.6 |
| R28C73 | MAI | TEH 2.8 |
| R29C73 | MAI | TEH 2.6 |
| R29C73 | MAI | TEH 2.9 |
| R32C73 | SAI | TEH 4.5 |
| R32C73 | SAI | TEH 16.7 |
| R32C73 | SAI | TEH 16.7 |
| R28C74 | SAI | TEH 2.6 |
| R28C74 | SAI | TEH 2.8 |
| R29C74 | MAI | TEH 2.4 |
| R29C74 | MAI | TEH 2.7 |
| R30C74 | MAI | TEH 2.7 |
| R30C74 | SAI | TEH 9.0 |
| R32C74 | MAI | TEH 2.9 |
| R32C74 | SAI | TEH 8.0 |
| R34C74 | MAI | TEH 2.8 |
| R34C74 | SAI | TEH 9.0 |
| R36C74 | MAI | TEH 2.7 |
| R36C74 | MAI | TEH 2.7 |
| R28C75 | MAI | TEH 2.6 |
| R28C75 | MAI | TEH 2.7 |
| R30C75 | SAI | TEH 6.0 |
| R30C75 | SAI | TEH 8.1 |
| R32C75 | SAI | TEH 2.8 |
| R32C75 | SAI | TEH 16.7 |
| R32C75 | SAI | TEH 16.9 |
| R33C75 | MAI | TEH 3.3 |
| R33C75 | SAI | TEH 5.3 |
| R35C75 | SAI | TEH 3.0 |
| R35C75 | SAI | TEH 5.0 |
| R24C76 | MAI | TEH 2.6 |
| R24C76 | MAI | TEH 2.8 |
| R13C77 | MAI | TEH 4.1 |
| R13C77 | MAI | TEH 7.5 |
| R16C77 | SAI | TEH 4.0 |
| R16C77 | SAI | TEH 8.6 |
| R16C77 | SAI | TEH 15.7 |
| R20C77 | SAI | TEH 2.9 |
| R20C77 | SAI | TEH 7.8 |
| R20C77 | SAI | TEH 10.4 |
| R25C77 | MAI | TEH 2.5 |
| R25C77 | MAI | TEH 2.6 |
| R30C77 | MAI | TEH 2.4 |
| R30C77 | MAI | TEH 2.8 |
| R4C78 | SAI | TEH 2.6 |
| R4C78 | SAI | TEH 2.8 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R8C78 | MAI | TEH 8.2 |
| R8C78 | SAI | TEH 8.8 |
| R17C78 | MAI | TEH 2.5 |
| R17C78 | SAI | TEH 6.8 |
| R17C78 | SAI | TEH 8.2 |
| R20C78 | SAI | TEH 2.6 |
| R20C78 | SAI | TEH 8.9 |
| R20C78 | SAI | TEH 10.0 |
| R24C78 | MAI | TEH 2.3 |
| R24C78 | MAI | TEH 2.5 |
| R29C78 | MAI | TEH 2.2 |
| R29C78 | SAI | TEH 2.3 |
| R30C78 | MAI | TEH 3.2 |
| R30C78 | SAI | TEH 15.7 |
| R13C79 | MAI | TEH 4.2 |
| R13C79 | SAI | TEH 16.0 |
| R13C80 | MAI | TEH 4.9 |
| R13C80 | SAI | TEH 7.7 |
| R16C80 | SAI | TEH 4.5 |
| R16C80 | SAI | TEH 4.9 |
| R17C80 | MAI | TEH 4.8 |
| R17C80 | MAI | TEH 16.7 |
| R17C80 | MAI | TEH 16.9 |
| R29C80 | MAI | TEH 2.0 |
| R29C80 | MAI | TEH 2.3 |
| R7C81 | SAI | TEH 9.1 |
| R7C81 | SAI | TEH 10.5 |
| R8C81 | SAI | TEH 7.0 |
| R8C81 | SAI | TEH 9.6 |
| R11C81 | SAI | TEH 2.3 |
| R11C81 | SAI | TEH 2.5 |
| R17C81 | SAI | TEH 5.4 |
| R17C81 | SAI | TEH 5.8 |
| R19C81 | MAI | TEH 2.3 |
| R19C81 | SAI | TEH 5.6 |
| R19C81 | SAI | TEH 5.8 |
| R19C81 | SAI | TEH 7.0 |
| R22C81 | MAI | TEH 2.3 |
| R22C81 | SAI | TEH 2.4 |
| R6C82 | SAI | TEH 7.2 |
| R6C82 | SAI | TEH 16.3 |
| R12C82 | SAI | TEH 2.5 |
| R12C82 | MAI | TEH 2.5 |
| R13C82 | MAI | TEH 5.8 |
| R13C82 | SAI | TEH 8.4 |
| R14C82 | SAI | TEH 4.5 |
| R14C82 | SAI | TEH 6.7 |
| R15C82 | SAI | TEH 6.6 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R20C82 | MAI | TEH 4.2 |
| R20C82 | SAI | TEH 9.7 |
| R22C82 | SAI | TEH 6.8 |
| R22C82 | SAI | TEH 7.8 |
| R25C82 | MAI | TEH 1.0 |
| R25C82 | MAI | TEH 4.2 |
| R25C82 | SAI | TEH 5.8 |
| R25C82 | SAI | TEH 5.9 |
| R6C83 | SAI | TEH 4.5 |
| R6C83 | SAI | TEH 16.0 |
| R9C83 | MAI | TEH 2.6 |
| R9C83 | MAI | TEH 2.6 |
| R10C83 | SAI | TEH 5.9 |
| R10C83 | SAI | TEH 6.0 |
| R11C83 | MAI | TEH 2.4 |
| R11C83 | MAI | TEH 2.4 |
| R13C83 | MAI | TEH 5.2 |
| R13C83 | SAI | TEH 16.1 |
| R17C83 | MAI | TEH 3.5 |
| R17C83 | SAI | TEH 16.0 |
| R19C83 | SAI | TEH 2.9 |
| R19C83 | MAI | TEH 7.5 |
| R19C83 | SAI | TEH 9.1 |
| R22C83 | SAI | TEH 3.1 |
| R22C83 | SAI | TEH 3.1 |
| R24C83 | MAI | TEH 2.3 |
| R24C83 | MAI | TEH 2.4 |
| R25C83 | SAI | TEH 2.6 |
| R25C83 | SAI | TEH 2.7 |
| R6C84 | SAI | TEH 5.8 |
| R6C84 | SAI | TEH 16.8 |
| R6C84 | SAI | TEH 17.1 |
| R9C84 | MAI | TEH 8.2 |
| R9C84 | SAI | TEH 10.7 |
| R10C84 | SAI | TEH 2.5 |
| R10C84 | SAI | TEH 2.5 |
| R17C84 | SAI | TEH 2.2 |
| R17C84 | SAI | TEH 2.3 |
| R18C84 | MAI | TEH 2.7 |
| R18C84 | SAI | TEH 2.9 |
| R18C84 | SAI | TEH 7.1 |
| R21C84 | MAI | TEH 2.2 |
| R21C84 | MAI | TEH 2.3 |
| R24C84 | MAI | TEH 2.2 |
| R24C84 | MAI | TEH 2.4 |
| R8C85 | SAI | TEH 4.7 |
| R8C85 | SAI | TEH 16.6 |
| R8C85 | MAI | TEH 16.9 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R9C85 | MAI | TEH 3.5 |
| R9C85 | SAI | TEH 12.0 |
| R11C85 | MAI | TEH 2.5 |
| R11C85 | MAI | TEH 2.7 |
| R12C85 | SAI | TEH 7.6 |
| R12C85 | SAI | TEH 7.9 |
| R10C86 | MAI | TEH 2.5 |
| R10C86 | MAI | TEH 2.5 |
| R14C86 | MAI | TEH 2.5 |
| R14C86 | SAI | TEH 2.9 |
| R14C86 | SAI | TEH 9.5 |
| R16C86 | MAI | TEH 2.5 |
| R16C86 | MAI | TEH 2.5 |
| R17C86 | MAI | TEH 1.8 |
| R17C86 | MAI | TEH 2.1 |
| R19C86 | MAI | TEH 2.3 |
| R19C86 | SAI | TEH 7.2 |
| R6C87 | SAI | TEH 5.2 |
| R6C87 | SAI | TEH 5.5 |
| R11C87 | MAI | TEH 2.4 |
| R11C87 | MAI | TEH 2.5 |
| R12C87 | MAI | TEH 2.2 |
| R12C87 | SAI | TEH 5.8 |
| R12C87 | SAI | TEH 6.8 |
| R12C87 | SAI | TEH 7.1 |
| R10C88 | SAI | TEH 2.5 |
| R10C88 | SAI | TEH 2.6 |
| R10C88 | SAI | TEH 3.9 |
| R11C88 | MAI | TEH 2.4 |
| R11C88 | MAI | TEH 2.4 |
| R13C88 | MAI | TEH 3.0 |
| R13C88 | MAI | TEH 5.3 |
| R13C88 | MAI | TEH 10.5 |
| R14C88 | SAI | TEH 2.4 |
| R14C88 | SAI | TEH 2.5 |
| R14C88 | SAI | TEH 2.8 |
| R16C88 | MAI | TEH 2.2 |
| R16C88 | MAI | TEH 2.8 |
| R17C88 | SAI | TEH 2.3 |
| R17C88 | MAI | TEH 2.6 |
| R17C88 | SAI | TEH 4.2 |
| R8C89 | SAI | TEH 2.7 |
| R8C89 | SAI | TEH 8.7 |
| R8C89 | SAI | TEH 9.0 |
| R9C89 | SAI | TEH 2.5 |
| R9C89 | MAI | TEH 2.6 |
| R12C89 | MAI | TEH 4.7 |
| R12C89 | MAI | TEH 12.7 |

| TUBES REROLLED IN STEAM GENERATOR A | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R14C89 | SAI | TEH 2.8 |
| R14C89 | SAI | TEH 10.0 |
| R9C90 | SAI | TEH 4.6 |
| R9C90 | SAI | TEH 5.0 |
| R13C90 | MAI | TEH 2.3 |
| R13C90 | SAI | TEH 2.7 |
| R3C91 | SAI | TEH 2.6 |
| R3C91 | SAI | TEH 11.8 |
| R3C91 | SAI | TEH 11.9 |
| R7C91 | SAI | TEH 4.9 |
| R7C91 | MAI | TEH 11.3 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R7C2 | MAI | TEH 3.1 |
| R7C2 | SAI | TEH 3.4 |
| R2C3 | MAI | TEH 6.1 |
| R2C3 | MAI | TEH 15.8 |
| R9C3 | SAI | TEH 3.5 |
| R9C3 | SAI | TEH 3.5 |
| R2C4 | SAI | TEH 5.4 |
| R2C4 | SAI | TEH 10.1 |
| R8C4 | SAI | TEH 7.1 |
| R8C4 | SAI | TEH 15.7 |
| R12C4 | MAI | TEH 2.9 |
| R12C4 | SAI | TEH 8.7 |
| R14C4 | MAI | TEH 0.9 |
| R14C4 | SAI | TEH 10.9 |
| R2C6 | MAI | TEH 6.9 |
| R2C6 | SAI | TEH 10.4 |
| R8C6 | MAI | TEH 2.5 |
| R8C6 | MAI | TEH 15.8 |
| R13C6 | SAI | TEH 4.2 |
| R13C6 | SAI | TEH 4.6 |
| R5C7 | MAI | TEH 7.4 |
| R5C7 | SAI | TEH 14.1 |
| R5C7 | SAI | TEH 14.1 |
| R9C7 | MAI | TEH 6.2 |
| R9C7 | SAI | TEH 6.9 |
| R12C7 | MAI | TEH 2.8 |
| R12C7 | SAI | TEH 15.9 |
| R15C7 | MAI | TEH 6.9 |
| R15C7 | SAI | TEH 9.0 |
| R19C7 | MAI | TEH 10.5 |
| R19C7 | SAI | TEH 11.0 |
| R20C7 | MAI | TEH 2.9 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R20C7 | MAI | TEH 15.9 |
| R5C8 | MAI | TEH 4.2 |
| R5C8 | MAI | TEH 15.6 |
| R7C8 | SAI | TEH 5.7 |
| R7C8 | SAI | TEH 5.8 |
| R8C8 | MAI | TEH 4.8 |
| R8C8 | SAI | TEH 7.5 |
| R14C8 | SAI | TEH 10.5 |
| R14C8 | SAI | TEH 10.8 |
| R16C8 | MAI | TEH 3.4 |
| R16C8 | MAI | TEH 15.9 |
| R5C9 | SAI | TEH 2.6 |
| R5C9 | MAI | TEH 15.9 |
| R8C9 | SAI | TEH 6.0 |
| R8C9 | SAI | TEH 6.1 |
| R12C9 | SAI | TEH 3.3 |
| R12C9 | MAI | TEH 15.7 |
| R2C10 | SAI | TEH 7.2 |
| R2C10 | SAI | TEH 7.5 |
| R8C10 | MAI | TEH 4.3 |
| R8C10 | MAI | TEH 15.8 |
| R16C10 | SAI | TEH 10.1 |
| R16C10 | SAI | TEH 10.2 |
| R17C10 | MAI | TEH 2.4 |
| R17C10 | MAI | TEH 16.0 |
| R20C10 | SAI | TEH 7.0 |
| R20C10 | SAI | TEH 7.0 |
| R6C11 | MAI | TEH 2.9 |
| R6C11 | MAI | TEH 15.9 |
| R16C11 | SAI | TEH 3.4 |
| R16C11 | MAI | TEH 15.8 |
| R19C11 | SAI | TEH 10.8 |
| R19C11 | SAI | TEH 15.8 |
| R20C11 | MAI | TEH 7.5 |
| R20C11 | SAI | TEH 9.4 |
| R23C11 | MAI | TEH 3.3 |
| R23C11 | SAI | TEH 4.0 |
| R8C12 | SAI | TEH 10.1 |
| R8C12 | SAI | TEH 10.2 |
| R16C12 | MAI | TEH 2.1 |
| R16C12 | MAI | TEH 16.0 |
| R18C12 | SAI | TEH 12.1 |
| R18C12 | SAI | TEH 12.3 |
| R16C13 | MAI | TEH 2.8 |
| R16C13 | SAI | TEH 9.7 |
| R26C13 | MAI | TEH 7.2 |
| R26C13 | SAI | TEH 10.1 |
| R28C13 | MAI | TEH 6.2 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R28C13 | MAI | TEH 15.8 |
| R29C13 | MAI | TEH 5.3 |
| R29C13 | SAI | TEH 8.6 |
| R17C14 | MAI | TEH 2.9 |
| R17C14 | SAI | TEH 8.1 |
| R23C14 | MAI | TEH 5.2 |
| R23C14 | MAI | TEH 15.8 |
| R26C14 | MAI | TEH 2.8 |
| R26C14 | SAI | TEH 12.1 |
| R17C15 | SAI | TEH 4.6 |
| R17C15 | SAI | TEH 5.0 |
| R22C15 | MAI | TEH 3.2 |
| R22C15 | SAI | TEH 15.8 |
| R25C15 | SAI | TEH 14.5 |
| R25C15 | SAI | TEH 14.7 |
| R27C15 | SAI | TEH 8.3 |
| R27C15 | SAI | TEH 8.5 |
| R29C15 | SAI | TEH 7.3 |
| R29C15 | SAI | TEH 15.9 |
| R20C16 | MAI | TEH 5.0 |
| R20C16 | SAI | TEH 11.3 |
| R22C16 | MAI | TEH 3.2 |
| R22C16 | SAI | TEH 3.3 |
| R28C16 | SAI | TEH 2.7 |
| R28C16 | SAI | TEH 2.8 |
| R33C18 | MAI | TEH 3.5 |
| R33C18 | SAI | TEH 9.2 |
| R23C19 | SAI | TEH 2.4 |
| R23C19 | MAI | TEH 4.0 |
| R23C19 | MAI | TEH 15.9 |
| R23C19 | SAI | TEH 16.0 |
| R26C19 | MAI | TEH 4.9 |
| R26C19 | SAI | TEH 8.0 |
| R30C19 | MAI | TEH 2.3 |
| R30C19 | SAI | TEH 15.7 |
| R32C19 | MAI | TEH 4.9 |
| R32C19 | MAI | TEH 16.0 |
| R30C22 | MAI | TEH 9.2 |
| R30C22 | MAI | TEH 15.6 |
| R32C22 | MAI | TEH 8.7 |
| R32C22 | SAI | TEH 10.4 |
| R33C23 | MAI | TEH 5.4 |
| R33C23 | SAI | TEH 8.3 |
| R1C26 | SAI | TEH 2.7 |
| R1C26 | SAI | TEH 15.9 |
| R37C26 | MAI | TEH 2.8 |
| R37C26 | SAI | TEH 6.4 |
| R1C30 | MAI | TEH 2.1 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R1C30 | SAI | TEH 15.5 |
| R33C30 | SAI | TEH 9.7 |
| R33C30 | SAI | TEH 9.8 |
| R37C30 | MAI | TEH 2.7 |
| R37C30 | SAI | TEH 3.9 |
| R41C30 | SAI | TEH 7.1 |
| R41C30 | SAI | TEH 11.0 |
| R33C32 | SAI | TEH 8.2 |
| R33C32 | SAI | TEH 8.7 |
| R37C32 | MAI | TEH 6.1 |
| R37C32 | MAI | TEH 11.0 |
| R33C33 | MAI | TEH 3.9 |
| R33C33 | SAI | TEH 15.7 |
| R37C33 | MAI | TEH 3.8 |
| R37C33 | SAI | TEH 10.9 |
| R39C35 | SAI | TEH 1.4 |
| R39C35 | SAI | TEH 10.7 |
| R33C36 | SAI | TEH 4.7 |
| R33C36 | SAI | TEH 10.3 |
| R38C36 | SAI | TEH 7.9 |
| R38C36 | SAI | TEH 8.0 |
| R34C37 | SAI | TEH 5.3 |
| R34C37 | MAI | TEH 15.9 |
| R1C39 | MAI | TEH 3.4 |
| R1C39 | SAI | TEH 15.3 |
| R34C40 | SAI | TEH 10.1 |
| R34C40 | SAI | TEH 13.1 |
| R41C40 | SAI | TEH 6.4 |
| R41C40 | SAI | TEH 7.4 |
| R41C40 | SAI | TEH 7.9 |
| R1C48 | MAI | TEH 2.9 |
| R1C48 | SAI | TEH 7.2 |
| R38C49 | MAI | TEH 6.1 |
| R38C49 | MAI | TEH 11.8 |
| R38C50 | MAI | TEH 5.6 |
| R38C50 | SAI | TEH 11.9 |
| R1C54 | SAI | TEH 3.1 |
| R1C54 | SAI | TEH 5.5 |
| R42C54 | SAI | TEH 5.0 |
| R42C54 | SAI | TEH 16.0 |
| R1C55 | MAI | TEH 3.6 |
| R1C55 | SAI | TEH 3.9 |
| R42C55 | SAI | TEH 5.5 |
| R42C55 | SAI | TEH 15.9 |
| R1C56 | MAI | TEH 2.5 |
| R1C56 | SAI | TEH 5.7 |
| R42C56 | SAI | TEH 7.3 |
| R42C56 | SAI | TEH 15.7 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R1C58 | MAI | TEH 2.5 |
| R1C58 | MAI | TEH 15.9 |
| R33C59 | MAI | TEH 4.7 |
| R33C59 | SAI | TEH 15.9 |
| R42C59 | MAI | TEH 7.4 |
| R42C59 | SAI | TEH 11.2 |
| R1C60 | MAI | TEH 3.0 |
| R1C60 | SAI | TEH 4.0 |
| R39C60 | MAI | TEH 6.4 |
| R39C60 | SAI | TEH 9.4 |
| R41C60 | SAI | TEH 6.8 |
| R41C60 | SAI | TEH 9.7 |
| R35C61 | SAI | TEH 7.9 |
| R35C61 | SAI | TEH 8.0 |
| R37C61 | SAI | TEH 5.3 |
| R37C61 | SAI | TEH 15.9 |
| R1C62 | SAI | TEH 3.6 |
| R1C62 | SAI | TEH 4.2 |
| R37C62 | MAI | TEH 7.3 |
| R37C62 | MAI | TEH 10.0 |
| R1C63 | SAI | TEH 3.6 |
| R1C63 | MAI | TEH 16.1 |
| R31C63 | MAI | TEH 5.3 |
| R31C63 | MAI | TH 15.9 |
| R32C63 | MAI | TEH 7.5 |
| R32C63 | SAI | TEH 13.0 |
| R36C63 | SAI | TEH 10.1 |
| R36C63 | SAI | TEH 10.6 |
| R37C63 | SAI | TEH 10.8 |
| R37C63 | SAI | TEH 11.0 |
| R39C63 | MAI | TEH 7.0 |
| R39C63 | MAI | TEH 11.8 |
| R41C63 | SAI | TEH 4.1 |
| R41C63 | SAI | TEH 4.7 |
| R1C64 | SAI | TEH 4.0 |
| R1C64 | SAI | TEH 5.9 |
| R1C64 | SAI | TEH 7.0 |
| R40C64 | SAI | TEH 6.3 |
| R40C64 | SAI | TEH 6.4 |
| R1C65 | MAI | TEH 2.7 |
| R1C65 | MAI | TEH 4.1 |
| R37C65 | MAI | TEH 6.0 |
| R37C65 | SAI | TEH 8.3 |
| R40C65 | SAI | TEH 4.8 |
| R40C65 | SAI | TEH 6.3 |
| R36C67 | SAI | TEH 5.2 |
| R36C67 | SAI | TEH 6.2 |
| R39C67 | MAI | TEH 6.8 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R39C67 | SAI | TEH 11.4 |
| R1C68 | MAI | TEH 2.4 |
| R1C68 | SAI | TEH 5.6 |
| R37C68 | MAI | TEH 7.0 |
| R37C68 | SAI | TEH 10.0 |
| R37C69 | MAI | TEH 8.2 |
| R37C69 | SAI | TEH 10.7 |
| R36C70 | SAI | TEH 14.9 |
| R36C70 | SAI | TEH 16.2 |
| R5C71 | MAI | TEH 5.9 |
| R5C71 | SAI | TEH 16.2 |
| R27C71 | SAI | TEH 4.6 |
| R27C71 | SAI | TEH 9.1 |
| R27C71 | SAI | TEH 10.1 |
| R37C71 | MAI | TEH 3.0 |
| R37C71 | SAI | TEH 9.0 |
| R24C72 | MAI | TEH 6.0 |
| R24C72 | SAI | TEH 10.7 |
| R36C72 | MAI | TEH 6.6 |
| R36C72 | SAI | TEH 10.7 |
| R2C73 | MAI | TEH 2.0 |
| R2C73 | MAI | TEH 2.2 |
| R27C73 | SAI | TEH 10.7 |
| R27C73 | SAI | TEH 10.7 |
| R32C73 | SAI | TEH 2.8 |
| R32C73 | SAI | TEH 3.5 |
| R17C74 | MAI | TEH 4.9 |
| R17C74 | SAI | TEH 9.4 |
| R22C74 | MAI | TEH 3.2 |
| R22C74 | SAI | TEH 16.4 |
| R7C75 | MAI | TEH 3.1 |
| R7C75 | MAI | TEH 16.0 |
| R21C75 | SAI | TEH 3.8 |
| R21C75 | SAI | TEH 3.9 |
| R22C75 | MAI | TEH 2.7 |
| R22C75 | SAI | TEH 11.6 |
| R24C75 | MAI | TEH 6.6 |
| R24C75 | MAI | TEH 15.8 |
| R27C75 | SAI | TEH 9.3 |
| R27C75 | SAI | TEH 11.1 |
| R7C76 | MAI | TEH 13.5 |
| R7C76 | SAI | TEH 15.7 |
| R16C76 | SAI | TEH 5.2 |
| R16C76 | SAI | TEH 15.9 |
| R22C76 | MAI | TEH 3.2 |
| R22C76 | MAI | TEH 15.9 |
| R24C76 | SAI | TEH 5.6 |
| R24C76 | MAI | TEH 15.9 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R26C76 | MAI | TEH 3.4 |
| R26C76 | SAI | TEH 6.5 |
| R27C76 | MAI | TEH 3.1 |
| R27C76 | MAI | TEH 11.9 |
| R29C76 | SAI | TEH 7.6 |
| R29C76 | SAI | TEH 7.7 |
| R14C77 | MAI | TEH 4.6 |
| R14C77 | SAI | TEH 5.9 |
| R16C77 | MAI | TEH 3.2 |
| R16C77 | SAI | TEH 15.8 |
| R17C77 | MAI | TEH 5.7 |
| R17C77 | MAI | TEH 11.4 |
| R27C77 | MAI | TEH 3.7 |
| R27C77 | MAI | TEH 9.8 |
| R1C78 | MAI | TEH 2.5 |
| R1C78 | SAI | TEH 15.9 |
| R13C78 | MAI | TEH 2.8 |
| R13C78 | SAI | TEH 16.0 |
| R14C78 | MAI | TEH 4.6 |
| R14C78 | SAI | TEH 7.3 |
| R18C78 | MAI | TEH 2.2 |
| R18C78 | MAI | TEH 2.4 |
| R22C78 | MAI | TEH 2.7 |
| R22C78 | MAI | TEH 4.2 |
| R26C78 | MAI | TEH 3.0 |
| R26C78 | SAI | TEH 15.5 |
| R13C79 | MAI | TEH 2.5 |
| R13C79 | MAI | TEH 15.8 |
| R22C79 | MAI | TEH 4.0 |
| R22C79 | SAI | TEH 4.4 |
| R9C80 | SAI | TEH 2.1 |
| R9C80 | SAI | TEH 2.4 |
| R18C80 | MAI | TEH 2.3 |
| R18C80 | MAI | TEH 2.4 |
| R19C80 | MAI | TEH 3.0 |
| R19C80 | SAI | TEH 9.8 |
| R19C80 | SAI | TEH 12.4 |
| R21C80 | MAI | TEH 7.8 |
| R21C80 | MAI | TEH 16.0 |
| R26C80 | SAI | TEH 3.4 |
| R26C80 | SAI | TEH 3.8 |
| R8C81 | MAI | TEH 3.1 |
| R8C81 | SAI | TEH 11.1 |
| R11C81 | SAI | TEH 2.4 |
| R11C81 | SAI | TEH 2.4 |
| R13C81 | MAI | TEH 3.0 |
| R13C81 | SAI | TEH 5.4 |
| R15C81 | SAI | TEH 4.5 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R15C81 | SAI | TEH 5.0 |
| R16C81 | MAI | TEH 3.3 |
| R16C81 | SAI | TEH 10.3 |
| R19C81 | MAI | TEH 2.7 |
| R19C81 | SAI | TEH 3.5 |
| R21C81 | MAI | TEH 7.5 |
| R21C81 | SAI | TEH 12.2 |
| R9C82 | MAI | TEH 3.8 |
| R9C82 | SAI | TEH 15.9 |
| R13C82 | MAI | TEH 7.1 |
| R13C82 | SAI | TEH 10.9 |
| R14C82 | SAI | TEH 6.0 |
| R14C82 | SAI | TEH 11.2 |
| R15C82 | SAI | TEH 2.9 |
| R15C82 | SAI | TEH 3.1 |
| R16C82 | MAI | TEH 3.1 |
| R16C82 | SAI | TEH 15.9 |
| R18C82 | SAI | TEH 2.4 |
| R18C82 | SAI | TEH 2.4 |
| R24C82 | SAI | TEH 9.5 |
| R24C82 | SAI | TEH 15.8 |
| R6C83 | MAI | TEH 0.1 |
| R6C83 | MAI | TEH 0.3 |
| R8C83 | MAI | TEH 0.2 |
| R8C83 | MAI | TEH 0.3 |
| R13C83 | MAI | TEH 2.8 |
| R13C83 | MAI | TEH 12.5 |
| R15C83 | MAI | TEH 1.7 |
| R15C83 | MAI | TEH 2.4 |
| R16C83 | MAI | TEH 2.8 |
| R16C83 | MAI | TEH 16.0 |
| R18C83 | MAI | TEH 2.4 |
| R18C83 | MAI | TEH 2.5 |
| R1C84 | MAI | TEH 5.8 |
| R1C84 | SAI | TEH 9.0 |
| R7C84 | SAI | TEH 4.9 |
| R7C84 | SAI | TEH 6.8 |
| R10C84 | SAI | TEH 13.2 |
| R10C84 | SAI | TEH 13.5 |
| R15C84 | SAI | TEH 1.8 |
| R15C84 | SAI | TEH 2.3 |
| R16C84 | MAI | TEH 6.3 |
| R16C84 | SAI | TEH 10.8 |
| R18C84 | SAI | TEH 8.4 |
| R18C84 | SAI | TEH 8.8 |
| R7C85 | MAI | TEH 4.5 |
| R7C85 | SAI | TEH 8.7 |
| R8C85 | MAI | TEH 2.9 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R8C85 | SAI | TEH 11.4 |
| R9C85 | SAI | TEH 2.9 |
| R9C85 | SAI | TEH 3.0 |
| R12C85 | MAI | TEH 1.9 |
| R12C85 | SAI | TEH 15.9 |
| R13C85 | MAI | TEH 6.9 |
| R13C85 | SAI | TEH 8.2 |
| R14C85 | MAI | TEH 3.8 |
| R14C85 | SAI | TEH 10.5 |
| R16C85 | SAI | TEH 7.6 |
| R16C85 | MAI | TEH 15.9 |
| R17C85 | SAI | TEH 3.6 |
| R17C85 | SAI | TEH 3.8 |
| R19C85 | MAI | TEH 2.4 |
| R19C85 | SAI | TEH 5.6 |
| R1C86 | SAI | TEH 5.9 |
| R1C86 | SAI | TEH 16.1 |
| R16C86 | MAI | TEH 3.6 |
| R16C86 | SAI | TEH 12.9 |
| R23C86 | SAI | TEH 2.7 |
| R23C86 | SAI | TEH 3.1 |
| R1C87 | SAI | TEH 6.8 |
| R1C87 | SAI | TEH 12.4 |
| R1C87 | SAI | TEH 16.0 |
| R4C87 | MAI | TEH 2.8 |
| R4C87 | MAI | TEH 10.5 |
| R4C87 | SAI | TEH 10.5 |
| R10C87 | SAI | TEH 7.3 |
| R10C87 | SAI | TEH 7.3 |
| R12C87 | SAI | TEH 2.9 |
| R12C87 | SAI | TEH 3.0 |
| R13C87 | MAI | TEH 3.0 |
| R13C87 | MAI | TEH 3.2 |
| R16C87 | MAI | TEH 3.0 |
| R16C87 | SAI | TEH 10.2 |
| R19C87 | SAI | TEH 3.0 |
| R19C87 | MAI | TEH 3.0 |
| R1C88 | MAI | TEH 2.8 |
| R1C88 | SAI | TEH 3.4 |
| R2C88 | SAI | TEH 2.3 |
| R2C88 | SAI | TEH 2.4 |
| R4C88 | MAI | TEH 3.7 |
| R4C88 | MAI | TEH 3.8 |
| R5C88 | SAI | TEH 7.7 |
| R5C88 | SAI | TEH 7.9 |
| R6C88 | SAI | TEH 3.0 |
| R6C88 | SAI | TEH 5.8 |
| R6C88 | SAI | TEH 11.0 |

| TUBES REROLLED IN STEAM GENERATOR B | | |
|---|---|---|
| TUBE | DEFECT | LOCATION |
| R6C88 | SAI | TEH 11.0 |
| R7C88 | MAI | TEH 4.2 |
| R7C88 | MAI | TEH 9.2 |
| R7C88 | SAI | TEH 15.8 |
| R8C88 | MAI | TEH 2.6 |
| R8C88 | SAI | TEH 9.7 |
| R13C88 | MAI | TEH 2.1 |
| R13C88 | MAI | TEH 2.5 |
| R17C88 | SAI | TEH 3.2 |
| R17C88 | SAI | TEH 3.6 |
| R4C90 | SAI | TEH 14.3 |
| R4C90 | SAI | TEH 16.0 |
| R8C90 | MAI | TEH 4.8 |
| R8C90 | SAI | TEH 15.4 |

## VIII. 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 valves or safety valves at normal operating pressure and temperature in 1995.

**Overpressure Protection During Low Pressure and Temperature Operation**

There were no challenges to the Unit 1 or Unit 2 power-operated relief valves during low temperature and low pressure operation in 1995.

## VIII. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications during operation of Unit 1 or Unit 2 in 1995 where reactor coolant activity exceeded that allowed by Technical Specifications.