Wisconsin Electric
A WISCONSIN ENERGY COMPANY

Point Beach Nuclear Plant
6610 Nuclear Rd.
Two Rivers, WI 54241
Phone 920 755-2321

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

DOCKETS 50-266 AND 50-301
1999 ANNUAL RESULTS AND DATA REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is the 1999Annual 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.

The report contains descriptions of facility changes, tests and experiments; personnel occupational exposures; steam generator in-service inspections; and commitment change evaluations that occurred during 1999.

Sincerely,

A. J. Cayia
Manager,
Regulatory Services & Licensing

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Enclosure

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

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WISCONSIN ELECTRIC

POWER COMPANY

ANNUAL RESULTS AND

DATA REPORT

1999

POINT BEACH NUCLEAR PLANT

UNITS 1 AND 2

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 and Data Report for 1999 is submitted in accordance with Point Beach Nuclear Plant, Units 1and 2, Technical Specification 15.6.9.1.B and filed under Dockets 50-266 and 50-301 for Facility Operating Licenses 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 1518.5 Mwt each. Each turbine-generator is capable of producing 515 Mwe (538 Mwe gross) of electrical power. The plant is located approximately ten miles north of Two Rivers, Wisconsin, on the west shore of Lake Michigan.

II. HIGHLIGHTS

UNIT 1

Highlights for the period January 1, 1999 through December 31, 1999, included: A power reduction on January 5 resulted because of a frozen common mini-recirculation flow line return to the refueling water storage tank for the safety injection and containment spray pumps (a notice of enforcement discretion was received and the line was subsequently returned to service); unit taken to hot shutdown on January 22 to repair Model 50-DH-350 breakers used with the 1P-1A&B reactor coolant pumps and 1&2P-28A&B main feedwater pumps; a 4 day maintenance outage in April to repair: An auxiliary feedwater pump recirculation line weld, El 66' containment hatch, main steam dump valves, W-3A control rod drive mechanism cooling fan, main generator disconnect, and blowdown system valves; repair of the No. 4 feedwater heater following a steam leak; and the unit began its twenty-fifth refueling outage on October 15, and was returned to service on December 9, 1999.

Unit 1 operated at an average capacity factor of 78.4% (MDC net) and an electrical/thermal efficiency of 34.8%. Unit and reactor availability were 80.7% and 81.9%, respectively. Unit 1 generated its 95 billionth kilowatt hour on January 29, 1999; its 96 billionth kilowatt hour on April 18, 1999; its 97 billionth kilowatt hour on July 21, 1999; and its 98 billionth kilowatt hour on October 8, 1999.

UNIT 2

Highlights for the period January 1, 1999 through December 31, 1999 included: Continuation of its twenty-third refueling/maintenance outage that began on December 5, 1998 with the generator placed online on March 7; power reduction to 20% for inspections of No. 4 feedwater heaters on May 14; repair of 2P-28A main feedwater pump on June 17; loss of Line 151 on June 24; and, repair of turbine EH governor valve #4 on September 11, 1999.

Unit 2 operated at an average capacity factor of 80% (MDC net) and an electrical/thermal efficiency of 35.1%. Unit and reactor availability were 82.1% and 84.4%, respectively. Unit 2 generated its 94 billionth kilowatt hour on March 28, 1999; its 95 billionth kilowatt hour on June 18, 1999; and its 96 billionth kilowatt hour on September 6, 1999.

III. AMENDMENTS TO FACILITY OPERATING LICENSES

During 1999 there were seven amendments issued by the U.S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Nuclear Plant Unit 1, and seven amendments issued to Facility Operating License DPR-27 for Point Beach Nuclear Plant Unit 2. License amendments are listed by date of issue and summarized below:

Amendment 186 to DPR-24; Amendment 191 to DPR-27, March 1, 1999: The amendments clarify the definition of refueling interval

Amendment 187 to DPR-24; Amendment 192 to DPR-27, March 2, 1999: The amendments remove the requirement for checking environmental monitors on a monthly basis.

Amendment 188 to DPR-24; Amendment 193 to DPR-27, March 2, 1999: The amendments change TS to ensure the 4kV bus undervoltage input to the reactor trip protective function is controlled in accordance with the design and licensing basis.

Amendment 189 to DPR-24; Amendment 194 to DPR-27, April 23, 1999: The amendments provide specific numerical setpoint limits for reactor trip, reactor coolant pump trip and auxiliary feedwater initiation on loss of power to the 4kV buses.

Amendment 190 to DPR-24; Amendment 195 to DPR-27, August 11, 1999: The amendments reflect personnel title changes, increase minimum operating crew shift staffing, relocate the Manager's Supervisory Staff composition and functional requirements to owner-controlled documents, and revises the procedure review and approval process.

Amendment 191 to DPR-24; Amendment 196 to DPR-27, December 6, 1999: The amendments remove the test requirements for snubbers from the Technical Specifications. These requirements are maintained in the PBNP In-Service Inspection Program.

Amendment 192 to DPR-24; Amendment 197 to DPR-27, December 23, 1999:The amendments update references in the Technical Specifications with respect to the updated Final Safety Analysis Report (FSAR). The update reflects the relocation of the referenced information to the updated FSAR.

IV. 10 CFR 50.59 & 72.48 SAFETY EVALUATIONS

PROCEDURE CHANGES

The following procedure changes requiring safety evaluations (SEs) were implemented in 1999:

1. AOP-10A, Safe Shutdown – Local Control, Revision 25. (Permanent)

   To minimize spurious operation of the power operated relief valves (PORVs) or block valves in the event of a fire, this procedure revision directs operators to take positive action to open manually operated breakers to remove power from the PORV control circuits and to remove power from motor control centers (MCCs) 1B-42 and 2B-42.

   Summary of Safety Evaluation: A fire in the central area of PAB El. 26' (Fire Zone 187) could cause spurious operation of the PORVs (1&2RC-430) in either or both units. Cables for solenoid operated PORVs and their associated block valves (1&2RC-516) run into and through the El. 26' pipeway (part of Fire Zone 187) to reach the containment locations of these valves. The associated block valves (1&2RC-516) are powered from MCCs (1&2B-42) located on PAB El. 26' in the same fire area. A fire in the area could cause the block valves to fail as is in the open position, and a hot short on the PORV cables could cause the PORV to open. The normal method of preventing this spurious operation is to fail instrument air to containment which places the PORVs in their shut position. The method of failing instrument air to containment is to shut valves IA-3047 and IA-3048 and open an adjacent vent valve. The instrument air valves are located in the respective unit's El. 26' pipeway that can not be reached until the fire is extinguished.

   Proceduralized actions are being taken to reduce the consequences of a postulated fire in the area and to minimize the potential for a spurious operation. To reduce the likelihood and size of a fire in the area, fire rounds are conducted and transient combustibles are controlled in fire Zone 187. This allows timely extinguishing of fires around the pipeway entrance. The area immediately surrounding the instrument air valves is free of combustibles, thus the instrument air valves remain free of damage because of fire and can be operated as soon as entry into the pipeway can be accomplished.

   With the PORV powered from its normal source, a hot short on the positive conductor energizes the solenoid. With the manually operated breaker open, it requires simultaneous hot shorts to both a positive and negative source. This is a much less likely event; however, it is considered credible. Since the block valves are normally open, a power source failure would leave the valves open. Since MCCs 1&2B-42 are located in the fire zone, loss of power to the block valves is presumed to occur. The revision to AOP-10A removes power from MCCs 1&2B-42 after an attempt is made to shut the block valves to prevent spurious operation.

   Since MCCs 1&2B-42 are located in the fire zone, the safe shutdown analysis assumes that equipment powered from the MCCs is not available. Therefore, deenergizing the MCCs does not result in an accident not evaluated in the CLB and does not result in a malfunction of equipment not already described in the CLB. Similarly, the safety function of the PORVs is to remain shut. Maintaining fire rounds and limiting combustibles in the area limits the consequences of a fire and limits delay in failing instrument air to containment to ensure that the PORVs are shut. The change does not pose an unreviewed safety question (USQ) nor does it require a change to the technical specifications (TS.) (SE 99-070)

2. DG-M02, Westinghouse Piping Class Summary. (Permanent)
   DG-M03, Bechtel Piping Class Summary. (Permanent)
   DG-M04, Stone & Webster Piping Class Summary. (Permanent)

   Design guideline changes clarify that Westinghouse or Bechtel supplement requirements and USAS B31.1-1967 are the original code of construction and design basis for the spent fuel pool (SFP) cooling piping systems. In addition, the design guidelines document the acceptability of use of the Stone and Webster specification, for the purpose of procuring replacement parts for the existing ASME Section III components.

Summary of Safety Evaluation: Technical Specification Change Request (TSCR) #36 requested NRC permission to remove administrative restrictions imposed on fuel movement because of storage rack modifications, pending completion of a SFP heat exchanger upgrade modification. The TSCR also informed the NRC of the SFP system's ASME Section III construction. The NRC acknowledged the ASME Section III system description, repeating it in their April 18, 1978 SER. The modification also updated the FSAR, including the ASME Section III system description. Since PBNP's voluntary commitment to ASME Section III for the modification's (M-278) code of construction, the unique code classification complicates maintenance and modification activities, by introducing new Stone and Webster specifications, which overlap and conflict with original Westinghouse and Bechtel piping and component specification, complicating ASME Section XI reconciliation.

No physical changes or additional tests inspections are required to meet the original design basis requirements. The ASME Section XI classification of these lines are not affected. No interim conditions are applicable since the plant is not physically altered. The change does not pose a USQ nor does it require a change to the TS. (SE 99-028)

3. EM, Environmental Manual, Revision 14. (Permanent)

The change eliminates water collection from E-12, the discharge flume, for the Radiological Environmental Monitoring Program (REMP). The changes to the Environmental Manual (EM) are administrative/editorial necessitated by the reorganization of the NPBU, changes in the name of a State department, typographical errors, and wording changes to clarify meanings. Because the EPA cancelled their water sampling program, a duplicate water sample for the State at E-05 for shipment to the EPA is eliminated. E-01 replaces E-12 as the co-located water site for the State monitoring program.

Summary of Safety Evaluation: A decrease in the margin of safety does not occur as a result of these changes. The changes do not affect SSCs within or outside the physical structure of PBNP and have no control or accident mitigation function. The change does not pose a USQ nor does it require a change to the TS. (SE 99-136)

4. EOP-1.3, Transfer to Containment Sump Recirculation, Revision 21. (Temporary)

The change establishes a minimum recirculation flow path for 1P-15B via 1SI-829B, 1SI-829C and 1SI-829D.

Summary of Safety Evaluation: A minimum recirculation flow path was unavailable for a period of time for the Unit 1 SI pumps because the normal minimum recirculation line was frozen. A minimum recirculation flow path is required to ensure the SI pump does not overheat when deadheaded. The work plan establishes a minimum recirculation flowpath for 1P-15B via 1SI-829B, 1SI-829C and 1SI-829D. The new flowpath uses IT-01, "The SI High Flow Test Line." Valves 1SI-829B, 1SI-829C and 1SI-829D are normally shut. When the SI high flow test line is used, flow is throttled via 1SI-829C to various flowrates to measure and trend pump/system performance. The same valve, 1SI-829C, is used to throttle the test line flow to the required nominal minimum flow of 110 gpm and locked in position. Valves 1SI-829B and 1SI-829D are isolated open. Flowrate verification is conducted using both the installed instrumentation and a portable UT flowmeter. The flowpath components are used as designed and remain within their design limitations. During this time the normal minimum recirculation flowpath is made inoperable by tagging shut 1SI-876B. 1P-15A is also inoperable. EOP-1.3, "Transfer to Containment Sump Recirculation," is revised to shut 1SI-829D. The change does not pose a USQ nor does it require a change to the TS. (SE 99-002)

Summary of Safety Evaluation: Valves 1SI-829B and 1SI-829D are locked open. Shutting 1SI-829D in EOP-1.3 does not affect the ability or time required to transfer to containment sump recirculation. The time required to shut 1SI-829D is equivalent to the time required to shut 1SI-876B, based on operator experience. The change does not pose a USQ nor does it require a change to the TS. (SE 99-002-01)

5. Framatome 1217919A, Field Procedure and Operation Instructions for Installation of a Flexible Stabilizer in a Recirculating Steam Generator, Revision 9. (Permanent)

The vendor procedure describes the method for installing flexible stabilizers in steam generator (SG) tubing. Stabilizers are installed when the potential exists for SG tube wear. The stabilizer consists of a 130.5" long stainless cable with Alloy 690 connecting pieces that allow it to connect to a standard Alloy 690 rolled plug. In addition, an Alloy 690 sleeve is crimped to the outside diameter of the cable to provide additional wear surface at the secondary fact of the tubesheet. Stabilizers are installed in tubes that are plugged when the potential exists for wear at the top of the tubesheet (e. g., loose part is present that can not be removed.) If a loose part can not be retrieved, additional evaluation of the loose part and the tube is required prior to plant operation.

Summary of Safety Evaluation: Installation of flexible stabilizers is performed in conjunction with SG tube plugging when the potential exists for wear of the tube to occur at the top of the tubesheet. The stabilizers have an Alloy 690 sleeve positioned at the top of the tubesheet to provide additional wear surface. The stabilizers connect to the top of the plug in the affected leg. Once installed, the stabilizers are fully contained within the plugged tube that were removed from service. Installation of the stabilizer is performed remotely using standard tube plugging equipment. Full foreign material exclusion practices are implemented. The effect of stabilizers on the reactor coolant system (RCS) is identical to tube plugging. Allowances are made in the FSAR and Technical Specifications (TS) for SG tube plugging to account for aging of the SGs. Plugging is performed using approved procedures and the installed plug was designed to the design requirements of the RCS pressure boundary. Key installation parameters are verified to ensure structural integrity and leak tightness of each tube plug. The evaluation addresses installation of the stabilizers on a preventive basis when potential loose parts are identified during eddy current inspections, when retrievability can not be assessed. Stabilizers installed when a loose part is to remain in the SG needs to be evaluated on a case-by-case basis. The change does not pose a USQ nor does it require a change to the TS. (SE 99-003)

6. NP 3.2.3, Secondary Water Chemistry Monitoring Program, Revision 7. (Permanent)

The change involves the action levels for feedwater pH and hydrazine. The pH action level is raised from pH 9.7 to 9.8 and the hydrazine action level for feedwater is raised from ≥5 ppb to ≥8 times the condensate pump discharge oxygen concentration or >20 ppb. FSAR Table 10.1-3 is changed by substituting "oxygen scavenger" for "hydrazine," and removes its numerical value. Instead, a footnote references NP 3.2.3, "Secondary Water Chemistry Monitoring Program."

Summary of Safety Evaluation: The change in pH accommodates the increase in hydrazine in accordance with the EPRI Secondary Water Chemistry Guidelines, Revision 4. The higher hydrazine levels protect piping from flow accelerated corrosion.

The increase in hydrazine creates more ammonia in the secondary system as the hydrazine breaks down. Ammonia can be detrimental to copper components. However, PBNP has very little copper in the secondary system. The only copper components are the condenser tube sheet and the hydrogen coolers. This equipment is kept at cool temperatures (approximately 80-100°F), thus decreasing the corrosion of copper. Research has shown that ammonia is not corrosive in low oxygen or low ammonia environments. The increase in hydrazine increases the condensate ammonia concentration from approximately 1.10 ppm to 1.99 ppm. Therefore, the increase in hydrazine does not harm the condenser tube sheet or the hydrogen cooler. The increase in feedwater pH has minor effects on the secondary system. EPRI TR-102952, "Advanced Amine Application Guidelines," Revision 2, shows that the iron solubility is lowest with a pH (25°C) between approximately 9.75 to 10.0. Therefore, increasing the pH from 9.7 to 9.8 does not increase the iron corrosion rate. The change does not pose a USQ nor does it require a change to the TS. (SE 99-089)

7. NP 8.4.17, PBNP Flooding Barrier Control, Revision 0. (New Procedure)

The new procedure creates a PBNP flooding barrier control procedure. It applies to doors and subsoil drain covers in the control building that were identified as needed, to reduce the effects of internal flooding. These items are either specifically described in the current licensing basis (CLB) or are important in minimizing the probability of core damage as evaluated in the PBNP probabilistic risk assessment (PRA). The procedure identifies the flood barriers covered by this procedure. It also defines a maximum inoperable time for non-CLB flood barriers without compensatory measure, and provides guidance on what makes a flood barrier door operable. The procedure provides acceptable compensatory measures when a CLB flood barrier is inoperable, or if a non-CLB barrier is inoperable beyond the allowed out of service time. It also provides recommended "not to exceed" door gaps for flood doors as required.

Summary of Safety Evaluation: The doors serve to mitigate the consequences in the event there is flooding in the turbine building from a circulating water pipe/expansion joint break (e.g., worst case break). The flooding could cause auxiliary feedwater pumps (AFPs), emergency diesel generators (EDGs), or 4160 Vac vital switchgear to malfunction, if flood barriers are not in place. The CLB for flooding is defined in NRC letter dated November 20, 1975. The three sets of doors discussed in the CLB are the AFP room doors, the vital switchgear room doors and the EDG room doors. To reduce the possibility of disabling safety-related equipment because of flooding while the doors are inoperable, the procedure requires an appropriate compensatory measure to be implemented for doors identified in the CLB. The procedural control of flood barriers does not impact sources of flood water, the capability to isolate flood sources, nor diverts flood water to other equipment important to safety. The change does not pose a USQ nor does it require a change to the TS. (SE 99-114)

8. ODCM, Offsite Dose Calculation Manual, Revision 12. (Permanent)

The ODCM change includes the addition of new dose factors, corrects dose factors, adds new descriptive material for dose factor calculations, and updates the administrative process of effluent quantification.

Summary of Safety Evaluation: The only ODCM change with the potential for affecting accident scenarios is the calculation of RMS setpoints. However, this ODCM section is not being changed. Therefore, the changes do not affect SSCs important to safety and can not create a new accident; increase the possibility of a previously analyzed accident; increase the consequences of a previously analyzed accident, event, or malfunction; and create the possibility of a malfunction of equipment important to safety. The margin of safety is not affected.

The ODCM changes do not diminish the ability to control effluents. The changes do not change our compliance with public dose limits as required by 10 CFR 20.1302, 10 CFR 50.36a, Appendix I of 10 CFR 50, or 40 CFR 190. The change does not pose a USQ nor does it require a change to the TS. (SE 99-004)

9. OI 152, HX-012A&B Component Cooling System Heat Exchanger Data Collection, Revision 0. (New Procedure)

The new procedure configures the plant to allow the collection of flow and temperature data on the HX-012A&B component cooling water heat exchangers. The collected data is used to ensure that HX-012A&B capacity is adequate to remove the required heat load at design limiting conditions.

Summary of Safety Evaluation: Highest heat loads for the component cooling (CC) heat exchangers occur during unit shutdown when the residual heat removal (RHR) system is first placed on-line. During this phase of plant operation and during post-LOCA plant operation, the CC heat exchangers help to remove decay heat from the RHR system. One CC heat exchanger must be able to remove the decay heat for the safe shutdown of one unit, and data collected during OI 152 is used to ensure that HX-012A&B can perform that function. The Train A PAB battery and inverter room ventilation system is isolated during performance of this test. In addition, CC heat exchanger HX-012A or HX-012B is isolated, depending on which heat exchanger is monitored. OI 152 involves data collection from non-intrusive UT flowmeters and externally mounted RTDs.

The plant is operated in a similar manner as during a normal unit shutdown for this test. The only changes are the isolation of the Unit 1 CC heat exchanger, and the isolation of Train A of the PAB battery and inverter room ventilation system. Both the CC heat exchangers and the PAB battery room ventilation system can be unisolated during the performance of OI 152 if needed. Heat removal from the CC system is not reduced by passing the CC flow for a unit through one heat exchanger, and higher flow rates within the heat exchanger do not cause adverse effects. Only one train of the PAB battery and inverter room ventilation system is normally in operation at any time and one train is able to adequately cool the PAB battery and inverter rooms. The test instrumentation is non-intrusive and adds negligible loads to the piping. The change does not pose a USQ nor does it require a change to the TS. (SE 99-115)

10. OM 3.27, Control of Fire Protection & Appendix R Safe Shutdown Equipment, Revision 8. (Permanent)

    The procedure change adds guidance for fire rounds and includes changes related to the Fire Protection Evaluation Report (FPER).

    Summary of Safety Evaluation: When both Units 1 and 2 have achieved and continue maintaining cold shutdown conditions, the step added on fire rounds provides guidance should hot shutdown components be suspended from these requirements. However, should plant conditions change, causing hot shutdown components (for either unit) to be made available or called upon to function, then at that time, the fire rounds that were once established for the respective hot shutdown components are reinstated immediately.

    The need for hot shutdown components is minimized when both units are at cold shutdown. Under these provisions, relief should be allowed from compensatory measures because both units have accomplished the ultimate goal by being in a safer plant mode.

    The TS for fire protection were relocated to the FPER. The FPER now provides administrative directions and guidance with regard to fire rounds related to Appendix R. The change does not pose a USQ nor does it require a change to the TS. (SE 99-044)

11. OP 4A, Filling and Venting Reactor Coolant System, Revision 44. (Permanent)

    SE 97-129 evaluated the temporary valve position of valves RC-525 and RC-522A to remain open during the next operating cycle to relieve a potential thermal overpressurization that could occur in the isolated pipe section as described in GL 96-06. This SE addresses the permanent valve position change of RC-525 and RC-522A and changes to FSAR Figures 4.2-1, Sheet 2 and 4.2-1A, Sheet 2.

    Summary of Safety Evaluation: The procedure and FSAR are revised to reflect valves RC-522A and RC-525 being in an open position during unit operation. This valve position prevents the RV level indication variable leg tubing from overpressurizing following a high energy line break or LOCA. The tubing is vented to containment atmosphere through LI-447B.

    The RV level transmitters and level indicator are normally isolated during unit operation and do not perform safety functions. Dual valve isolation from the RCS is maintained and no new flow paths are created for a loss of reactor coolant. The change is passive in nature and has no effect on safety-related equipment. The change does not pose a USQ nor does it require a change to the TS. (SE 97-129-01)

12. ORT 3A, (Unit 2,) Safety Injection (SI) Actuation with Loss of Engineered Safeguards AC (Train A), Revision 31. (Permanent)

    The revisions remove the Train B safeguards testing from ORT 3A. ORT 3A, Revision 30, contained both Train A and Train B safeguards testing. Train A was fully tested with selected Train B testing performed. ORT 3B, Revision 29, fully tested Train B with selected Train A testing performed. Those tests were scheduled such that ORT 3A and ORT 3B were alternated with refueling outages. Therefore, prior to this change, each train was fully tested every other outage. As a result of this change, ORT 3A (Train A) and ORT 3B (Train B) are performed during each refueling outage, thereby fully testing their associated engineered safeguards train.

Summary of Safety Evaluation: The test is performed with Unit 2 in cold or refueling shutdown and RCS temperature <140°F. Applicable action statements for both units are entered. G-02 and subsequently G-01 EDG are aligned to 2A-05 and a simultaneous Unit 2 Train A SI and 2A-05 undervoltage is initiated in accordance with TS 15.4.6.A.2. Automatically sequenced loads available on 2A-05 are loaded onto G-01 and G-02 EDG.

The testing configuration and operation of equipment associated with the test are within the system and component requirements specified in the CLB. The load carrying capabilities of G-01 and G-02 EDGs were demonstrated during the performance of PBTP 65, "G-01/G-02 Functional Test with U2 Accident Loads and U1 Cold Shutdown Loads." Cold shutdown requirements of Unit 2 are maintained by the train that is not being tested (Train B).

The revision meets the system and component requirements of the CLB for Unit 2 in a cold or refueling shutdown condition. Applicable action statement entries are made for Unit 1 operation. The change does not pose a USQ nor does it require a change to the TS. (SE 99-011)

13. ORT 3B, (Unit 2,) Safety Injection Actuation with Loss of Engineered Safeguards AC (Train B), Revision 30. (Permanent)

The change removes the Train A safeguards testing from ORT 3B. ORT 3B, Revision 29, contained both Train A and Train B safeguards testing. Train B was fully tested with selected Train A testing performed. ORT 3A, Revision 30, fully tested Train A with selected Train B testing performed. Those tests were scheduled such that ORT 3A and ORT 3B were alternated with refueling outages. Therefore, prior to this change, each train was fully tested every other outage. As a result of this change, ORT 3A (Train A) and ORT 3B (Train B) are performed during each refueling outage, thereby fully testing their associated engineered safeguards train.

Summary of Safety Evaluation: The test is performed with Unit 2 in cold or refueling shutdown and RCS temperature <140°F. Applicable action statements for both units are entered. G-04 and subsequently G-03 EDG are aligned to 2A-06 and a simultaneous Unit 2 Train B SI and 2A-06 undervoltage is initiated in accordance with TS 15.4.6.A.2. Automatically sequenced loads available on 2A-06 are loaded onto G-03 and G-04 EDGs.

The testing configuration and operation of equipment associated with the performance of the test are within the system and component requirements specified in the CLB. The capability of the equipment to perform under these loading scenarios was demonstrated by PBTP 66, "G-03/G-04 Functional Test with U2 accident loads and U1 cold shutdown loads." Cold shutdown requirements of Unit 2 are maintained by the train that is not being tested (Train A).

The revision meets the system and component requirements of the CLB for Unit 2 in a cold or refueling shutdown condition. Applicable action statement entries are made for Unit 1 operation. The change does not pose a USQ nor does it require a change to the TS. (SE 99-012)

14. PBPT-087, Control Room Pressure Boundary Test Procedure, Revision 0. (New Procedure)

The test checks the current leak tightness of the control room envelope, based on required filtered make-up air to maintain a positive pressure in the control room.

Summary of Safety Evaluation: The control room HVAC system provides heating, ventilation, air conditioning, and radiological habitability for the control room and computer room, which are within the control room envelope. The control room is maintained at a positive pressure to prevent unfiltered inleakage into the control room envelope. Special test procedure PBTP 087 checks the current leak tightness of the control room envelope, based on required filtered make-up air to maintain a positive pressure in the control room. The test configures the control room HVAC system to operate in a reconfigured emergency mode (Mode 3/4), with identified major inleakage/exfiltration paths isolated. The major inleakage/exfiltration paths are the snack room and toilet exhaust fan and the ventilation supply in the mechanical equipment room.

The test reconfigures the control room ventilation system into a modified emergency operating mode, wherein a portion of the recirculation air and the outside air make-up is filtered through the emergency filter train. The volume of the outside air make-up is varied to obtain a desired positive pressure in the control room envelope. The resultant variations occur in the pressure of the control room do not present the potential to affect or create an accident or event, nor do they detrimentally impact the performance of plant equipment. The ventilation exhaust from the snack room and control room toilet is isolated during this test. Since no significant change in the control room ventilation or cooling occurs, the isolation of the ventilation exhaust from the snack room and control room toilet do not impact control room operations and equipment. The ventilation supply to the control building mechanical equipment room is also isolated during this test. This results in a loss of cooling and ventilation within the mechanical equipment room. Mechanical equipment room temperatures are monitored during the test to ensure that required environmental conditions for equipment within this room are not exceeded. Prior evaluations took credit for positive pressure in the mechanical equipment room. By removing the ventilation supply from this room, where the emergency filter train is located, the potential exists for additional bypass leakage into the control room envelope. Consistent with this potential impact on the filter bypass, the test is performed in accordance with the requirements of those action statements stated in TS 15.3.12.3. The control room ventilation system is restored to its normal operating configuration (Mode 1) as described in the CLB.

The TS address the efficiency and testing of the HEPA filters and charcoal adsorbant trays in the control room HVAC system. The test is performed in accordance with TS; therefore, the change does not reduce the margin of safety of the control room HVAC system, the control room envelope, the protections afforded control room personnel, or the plant. The change does not pose a USQ nor does it require a change to the TS. (SE 99-098)

15. PBTP 098, Alignment for and Recovery From 1A52-57, 1A-05 Normal Supply Breaker Maintenance, Revision 0. (New Procedure)

The new procedure places the plant in a condition that allows inspection and repair of the 1A-05 normal supply breaker, 1A52-57. During the breaker maintenance activity, G-01 EDG is used to supply power to 1A-05. TS 15.3.7.A.1.i and 15.3.7.B.1.f relate to the use of EDGs for supplying power to the 1A-05 bus when a unit is critical. Unit 1 is in a hot shutdown condition and Unit 2 defueled during performance of the test. TS 15.3.7.B.1.f is applicable during those plant conditions since the action statement was entered while Unit 1 was at-power because of concerns regarding 1A52-57 capability to open and G-01 EDG supply breaker 1A52-60 to close. The concern regarding the G-01 EDG supply breaker to 1A-05 was resolved. The procedure transfers loading from 1A-03 to G-01 EDG to allow the 1A52-57, 1A-05 normal supply breaker to be opened for inspection and maintenance. It also will reclose 1A52-57, 1A-05 normal supply breaker to allow G-01 EDG to be returned to its normal standby status and exit TS 15.3.7.B.1.f LCO action statement.

Summary of Safety Evaluation: The G-01 EDG voltage regulator is initially in manual during this test when the normal supply breaker, 1A52-57 is opened. However, initial bus voltage is approximately 4300 V, with 6% droop; full load voltage (assuming no operator action) is greater than 4 kV. This is greater than the 3937 V degraded grid voltage relay minimum setpoint as delineated in TS. This assures operability of safety-related components. Normally, during the performance of G-01 EDG surveillance testing, the 1B-03 undervoltage load stripping is maintained by operating test switches at the G-01 EDG supply breaker. During this test, these test switches are not repositioned. A loss of offsite power would cause a G-01 EDG overload. This results in a G-01 EDG breaker trip. This then enables the 1B-03 undervoltage stripping for subsequent resequencing of loads onto G-01 EDG, if necessary. This situation would exist for the brief period of time that both the G-01 and 1A-05 normal supply breakers are closed simultaneously. The test switches for 1B-03 load stripping are maintained in their normal accident mode configuration. Therefore, the performance of the test does not create the possibility of an accident or event of a different type than previously evaluated in the CLB.

Equipment is operated within normal design parameters. Therefore, performance of this procedure does not reduce the margin of safety defined in the basis of TS. The change does not pose a USQ nor does it require a change to the TS. (SE 99-013)

16. PBTP 099, 1SI-12X (4-8) Contact Test, Revision 0. (New Procedure)
PBTP 100, 1SI-23X (3-7) Contact Test, Revision 0. (New Procedure)

The procedures test spare contacts on relays 1SI-12X (Train A) and 1SI-23X (Train B) respectively, which are located in the safeguards relay racks. The spare contacts are wired to terminal strips along the side of the safeguards relay racks. The contacts are tested by depressing the plunger to manually actuate the relay and check for continuity at the terminal strips. The testing supports installation of MR 97-081 that converts the original manual service water (SW) valves on the inlet of the spent fuel pool (SFP) heat exchangers to motor-operated valves (MOVs) that isolate upon a SI signal from either unit. PBTP 100 tests the Unit 1 Train A safeguards relay contact. PBTP-099 tests the Unit 1 Train B safeguards relay contact.

Summary of Safety Evaluation: The Train A relay performs the following safeguards functions: 1) Trips both main feedwater pumps; 2) Provides an open signal to the Train A SI accumulator outlet MOV (normally open with power isolated to it); 3) Provides an open signal to the Train A low head SI core deluge isolation valve; 4) Provides an open signal to the SI pump Loop A injection valve; and, 5) Defeats the Unit 2 4160 V fast bus transfer scheme. Testing occurs with the unit shutdown and RCS ≤350°F, SI blocked with the Train A SI pump tagged out, and both main feedwater pumps secured. The accumulator outlet MOV is not repositioned during the test because the normal alignment for these plant conditions is having the valve shut with its power supply isolated. This is verified prior to the test. The normal position for the SI pump Loop A injection valve is open; therefore, an open signal generated by the test does not reposition the valve. It is verified that the valve is open prior to the test. The RHR system valve LCO (TS 15.3.3.A.3.c) is entered and the Train A low head SI core deluge isolation valve is allowed to stroke during the test. The operators are aware of the affect this testing has on the 4160 V fast bus transfer scheme for Unit 2. There is no action statement associated with this circuitry.

The Train B relay performs the following safeguards functions: 1) Trips the electric AFP, P-38B; 2) Provides the SI input to the degraded grid scheme; 3) Provides an open signal to the containment recirculation heat exchanger emergency flow control valve; 4) Feeds alarms and indication in the control room. Testing per PBTP 99 is performed with the unit shutdown and the RCS ≤350°F, SI blocked, and the Train B SI pump tagged out. The electric AFP action statement (TS 15.3.4.C.2) is entered for Unit 2 and P-38B AFP is secured for the test. The containment accident fan cooler (CFC) action statement (TS 15.3.3.B.2.c) is entered for Unit 1 and power is isolated to the containment recirculation heat exchanger emergency flow control valve in order to prevent it from operating. The degraded grid circuitry is not disabled during the test; therefore, no action statement entry is required associated with normal or emergency power.

Relays in the safeguards relay cabinets actuate safeguards equipment in response to an accident. The equipment actuated by the safeguards racks is not an accident initiator. The safeguards circuitry does initiate a reactor trip, which is an event evaluated in the CLB. The testing is performed with the unit shutdown to eliminate the possibility of initiating a reactor trip while performing the test. There are no Unit 2 accidents that can be initiated from the Unit 1 safeguards relay cabinets. Therefore, the tests do not increase the probability of accidents. Plant conditions are set in the initial conditions and the appropriate required action entered for safety-related equipment. The testing is completed in small portions within that allowed by the required action. Redundant equipment is verified operable prior to entering the required action. Based on this, no equipment important to safety is adversely affected by the tests and the consequences of an accident or event is not increased. The change does not pose a USQ nor does it require a change to the TS. (SE 99-040)

17. RESP 4.3, Bank Worth Measurement by Dynamic Insertion, Revision 0. (New Procedure)

The procedure performs measurement of control bank worth. During the test, the reactor is critical below the point of adding heat. The RCS is at normal operating pressure and no load temperature. Power is limited to the requirements of low power physics testing exemptions. Required safety systems are operable. The bank measured is inserted in a continuous motion from the all rods out (ARO) position to near fully inserted (two steps) as indicated on the bank demand counters. The flux signals from the upper and lower section of one power range nuclear instrument channel is recorded by the advanced digital reactivity computer (ADRC) while the bank is inserted. The bank is then withdrawn to the ARO position and the remaining banks are measured similarly. The measurement requires an adjustment to the rod control system during the test to allow the banks to move at maximum speed (~68 steps per minute) in the bank select mode. The bank is inserted in one continuous motion. Failure to adhere to this requirement requires remeasurement of the bank.

Summary of Safety Evaluation: The rod control and chemical and volume control systems (CVCS) are not affected by the new technique. SER 91-099 evaluated the method. It was reviewed and determined that no changes in plant systems, administrative requirements or the CLB change the conclusions of that report. The report remains valid.

No discussion of the CVCS malfunction was contained in that SE. This SE finds that there is no increase in the probability or consequences of that malfunction caused by the use of this new procedure. Also, a reactor trip caused by the source range unblocking and increase to the trip setpoint is prevented by procedural controls contained in RESP 4.3. The change does not pose a USQ nor does it require a change to the TS. (SE 99-014)

18. RMP 9225-1, Containment Air Lock Door Alarm Defeating and Restoration, Revision 0. (New Procedure)

    The new procedure provides direction for defeating and restoring personnel airlock door alarms. The alarms are defeated to reduce control room operator distraction from actual alarms during conditions when traffic in and out of containment is high and the doors are closed. Defeating the door alarms does not result in the airlock door being unable to be closed and sealed. The door alarms are not required for operability of the containment airlock doors. The FSAR presently contains the statement, "An audible alarm sounds in the control room when the outer door of the containment is opened." The FSAR does not reflect the purpose of the door alarm to remotely monitor containment access and a potential breach of containment from the control room.

    Summary of Safety Evaluation: The only analyzed event in cold or refueling shutdown that requires the containment airlock doors to be closed is a fuel handling accident inside containment. The doors are closed in response to the accident in accordance with AOP-8B and subsequently limit release of airborne radioactivity to the environment to the same conditions as analyzed in the CLB. When the plant is above cold shutdown, containment integrity is set and the airlocks are maintained and controlled by TS requirements. Containment entry above cold shutdown is controlled by the radiological work permit process. Containment integrity is assumed in the accident analysis for control of radioactive releases and is controlled when operating above cold shutdown. Operation of the airlock doors is performed by personnel trained in proper operation of the doors to maintain containment integrity.

    There is no increase in the probability of credible accidents or events or new events that could be created. The alarm limit switches are an attachment to the containment airlock doors for a basic indication of door position. The defeating of the door limit switch does not affect the operation or the sealing capability of the airlock doors. There are no events described in the CLB initiated by malfunction of the containment airlock doors or the door limit switches. The change does not pose a USQ nor does it require a change to the TS. (SE 99-104)

19. RMP 9367, Containment Coatings, Revision 3. (Permanent)

    The procedure revision allows a new coating system on concrete walls and floors inside containment. It also allows an old system to be reused under stricter restrictions. The surface preparation and application sections were rewritten to include painting on concrete walls and floors in addition to shortening steps. The data sheets were changed for ease of use by personnel. Applicable sections from NP 8.4.14 are included and new forms created to help track the unqualified coatings applied during coating activities.

Summary of Safety Evaluation: The new coating system for concrete walls and floors was tested in a conservative DBA environment and is Service Level 1 QA approved. The coating does not delaminate from the surface during a DBA and cause the containment sump to clog. Therefore, the emergency core cooling system remains operable and does not increase the radiological consequences of an accident. The change does not pose a USQ nor does it require a change to the TS. (SE 99-096)

20. RP 1A,Preparation for Refueling, Revision 51. (Permanent)
RMP 9096, Reactor Vessel Head Removal and Installation, Revision 20. (Permanent)

The change relates to a commitment change that allows a revision to RP 1A and RMP 9096. These procedures contain the requirements for personnel entering the keyway to check for cavity seal, NIS seal, or sandbox cover leakage (hereon called seal leakage). The reactor vessel (RV) head lift sequence is revised to use sump "A" monitoring to check for seal leakage.

Summary of Safety Evaluation: The change involves a revision of the RV head lift sequence by changing the CLB as defined in response to NRC Bulletin 84-03, "Refueling Cavity Water Seal." The bulletin response committed to send personnel into the keyway after the reactor cavity seal was installed and refueling water is flooded just over the seal to inspect for seal leakage. RP 1A and RMP 9096 require that personnel be sent into the keyway to check for cavity seal leakage. Based on past operating experience, a radiological savings to the personnel, and the available safety measures in place, that the requirement to send personnel into the keyway be removed and sump "A" be used to check for seal leakage.

The change does not increase the probability of an accident or event because failure or leakage of the seal is independent of inspection (e.g., looking at the seal will not prevent its failure.) The inspection only serves to initially determine if there is leakage, its location, and whether or not the leakage is too significant to proceed with floodup. After floodup has commenced and the thimbles are retracted, personnel cannot enter the keyway and leakage detection becomes a matter of monitoring sump levels in the control room. There is a possibility that there could be increased corrosion to the containment liner if there was undetected leakage. The amount would be minimal because current procedure allows for some leakage and a large amount of leakage would require the refueling cavity to be drained and the seal fixed. The change offers a reduction in radiological exposure because personnel are not sent in the keyway. Also, should there be a gross seal failure while personnel are in the keyway it would prevent the possibility that they could be exposed to a large amount of contaminated refueling water. The change does not introduce other accidents or events other than the one analyzed in the bulletin response. Containment measures are taken to prevent malfunctions of equipment because of seal leakage or failure because the keyway would be the first place to flood in accidents involving the reactor or during refueling. The water from the seal leakage can be returned directly to the RV through sump "B" and the core deluge lines. Finally, no TS exist for the seal ring or inspection of the keyway by personnel. The change does not pose a USQ nor does it require a change to the TS. (SE 99-113)

21. RP 8, Unloading the Multi-Assembly Sealed Basket (MSB), Revision 8. (Permanent)

The change involves using a smaller thickness material for the storage sleeves for the fuel assemblies in the multi-sealed basket (MSB) than analyzed in the independent spent fuel storage installation (ISFSI) Safety Analysis Report (SAR). The effects were previously analyzed in SER 95-079, SER 95-079-01, and SE 98-0140, but a recent re-examination of only the criticality aspects of the change (looking at the criticality issue during the loading process when the loaded ventilated storage cask (VSC) is flooded with water), examining the change while assuming unborated water vice borated water is used, prompted this re-evaluation against the criteria of 10 CFR 72.48.

10 CFR 72.48 Evaluation Summary: The examination concluded that the effect of using unborated water in the criticality examination of the change creates a higher K_{eff}<0.95, as a measure of the reactivity existing in the MSB during the dynamic loading configuration, than previously calculated. The examination also, however,

goes on to look at the approach to the licensing basis-stipulated margin to criticality of keeping K_{eff}<0.95, and concludes, for the VSCs/MSBs affected by the change, in consideration of the characteristics of the fuel loaded or planned to be loaded, that the margin of safety represented by the 0.95 figure is maintained. The change does not pose a USQ nor does it require a change to the Certificate of Compliance (C of C). (SE 99-006)

22. 0-SOP-Y-Y15, Y-15 and Y-16 120 V Instrument Bus Alternate Source Distribution Panels, Revision 0. (New Procedure)

The new procedure provides detailed instructions for de-energizing and re-energizing 120 V instrument panels Y-15 and/or Y-16 (both are ultimately powered from alternate shutdown transformer X-08; Y-16 is powered from Y-15).

Summary of Safety Evaluation: Loss of inverter output and subsequent deenergization of an instrument channel is the worst case condition that can occur during this procedure. Since the loss of an instrument channel is already a design condition, removing the backup power is bounded by current accident analysis.

The procedure affects the 120 VAC vital instrument inverters that supply the vital buses (addressed in TS 15.3.7) but it does not render the inverters or the buses inoperable because only the backup power source is removed. The procedure does not conflict with, or require entry into TS action statements. Protection is still provided for fault conditions and an alternate inverter is still available. The change does not pose a USQ nor does it require a change to the TS. (SE 99-066)

MODIFICATIONS

The following modifications were implemented in 1999:

1. MR 88-177*A, (Unit 1), Purge, Supply and Exhaust.

   The modification improves the reliability and performance of the existing control system for the outboard purge supply and exhaust containment isolation valves, 1VNPSE-3212 and 1VNPSE-3244. It also addresses a 20 minute hold time for throttling valve 1VNPSE-3212 by regulating air pressure to its actuator.

   Summary of Safety Evaluation: The pneumatic control system uses three active components (solenoid, regulator and a check valve) and no jumpers. The 33/AC contacts on the limit switch for 1VNPSE-3212 are switched to an unused set of 33/AC contacts. The new control system is installed during Unit 1 Refueling 25 (U1R25). The outboard valves are shut during installation; however, the boot seals are not pressurized. Also, the instrument air to panel RK-41, PAB ventilation/Unit 1 containment purge instrument rack, is isolated because of the new instrument air tie-in. These conditions disable the containment purge ventilation and the PAB supply fan, W-35. Containment has a minimum temperature requirement of 50°F and purge system supplies heating and cooling to containment. The moderate temperatures during U1R25 do not challenge the minimal temperature requirement. The PAB exhaust fans are available to maintain the PAB at a slightly negative pressure. The reactor is in cold shutdown during installation and disabling these systems during shutdown is within the licensing basis.

   Valves 1VNPSE-3212 and 1VNPSE-3244 fail shut and are require to be shut for containment integrity. Failure of these valves' control system could deflate the valve boot seal, but this can not initiate an accident or event. Failure of the instrument air system could cause a reactor trip and challenge safety systems, but failure of the instrument air system is not increased by this modification. The modification enhances the reliability of the instrument air system local to the purge valves by significantly reducing the amount of tubing, the number of active components, and by replacing the copper tubing with heavier wall stainless steel tubing. The modification meets the design basis stroke times, leak rate requirements and seismic requirements. The replacement components meet or exceed the requirements of the original Bechtel Specification 6118-M-189 and the requirements of the design Code AWWA C-504-66. Switching the limit switch contacts in control circuit for 1VNPSE-3212 does not change the response of the valve to a containment ventilation isolation signal, but does reduce the time that the containment ventilation isolation signal is bypassed. With the control switch held in the open position, the containment isolation signal for 1VNPSE-3212 is bypassed. The control logic accepts a close signal from the automatic position. After the modification, the control switch is held in the open position for less time. Each valve has an air accumulator that provides several hours of boot seal make-up air. The reliability and leak rate of the boot seal air is tested quarterly, and the acceptance criteria is not changed by the modification. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-072)

   Summary of Safety Evaluation: The primary auxiliary building (PAB) exhaust fans are available to maintain the PAB at a slightly higher negative pressure. Switching the limit switch contacts in control circuit for 1VNPSE-3212 does not change the response of the valve to a containment ventilation isolation signal, but does not reduce the time that the signal is bypassed. The reliability and leak rate of the boot seal air is tested quarterly under the IST program. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-072-01)

2. MR 89-202*A and *B, (Unit 1 and 2), CVCS.

   The modification enlarges and provides for accessibility to the addition pot for 1&2T-05 CVCS chemical mixing tank. It also removes a tripping hazard that existed because of the current drain line location.

   Summary of Safety Evaluation: The work occurs during mid-cycle when chemicals are not normally required. The chemical mixing tank is made inoperable for approximately 2-3 days when the tank is lowered and moved as an improvement. The tank supports are relocated and the system restored. The additional funnel for the tank is replaced with a larger funnel to reduce the chance of spilling chemicals. The drain line for the tank is rerouted through the El. 26' floor so operators are able to see flow through the pipe. The drain line meets the pipe class of the drain. The sight glass also meets this pipe class requirement.

During the mid-cycle, chemicals are generally only required if a transient occurs in which the reactor coolant system (RCS) lithium concentration is diluted below the administrative control band. Lithium hydroxide is then added through the chemical mixing tank. Chemicals can not be added when 1&2T-05 are inoperable during this modification. If this should occur, the consequences are minor since the pH level would only decrease below the administrative band for a short period of time. This does not affect equipment reliability since it routinely experiences low lithium concentration during refueling shutdown. In addition, lithium and pH are not TS items. The El. 26' floor is not a flood barrier, and a drain is located almost directly below the location for a penetration. Furthermore, no equipment important to safety exists on the floor in close proximity to the penetration. Piping and new equipment are in compliance with ASME B31.1 and the current pipe class. The modification does not pose a USQ nor does it require a change to the TS. (SEs 99-031, 99-051)

3. MR 92-130, (Unit 1), Fuel Handling System.

The modification upgrades the 1Z-16 fuel manipulator control system. The new system is more reliable, easy to maintain, and includes several operational enhancements. Major features of the new system include: 1) A variable frequency drive replaces the original thyristor drive boards; 2) A programmable logic controller is used to control the system; 3) Interlocks designed into the current system are incorporated into the new design; 4) Separate jog control switches are provided for the trolley, bridge, and hoist; 5) A new weight monitoring system with digital readout allows more flexibility and additional interlocks; 6) The existing bridge position indication system is replaced with a more direct indication using a video camera and monitor; and, 7) A digital readout of the mast vertical position is provided.

Summary of Safety Evaluation: Installation occurs during U1R24 with the disconnect and/or removal of the components intended to be upgraded and/or replaced. The motor control center and control console are deenergized and the wires disconnected for the motor control center. The wires for the control console are bundled together using containment qualified fasteners and secured.

Fuel handling accidents inside containment are described in FSAR Section 14.2.1, with further analysis contained in WE letter dated March 9, 1977 regarding fuel handling accident in containment, and NRC SER dated June 20, 1979. These accidents are prevented through procedural adherence, supervisory oversight, and safety interlocks designed into the fuel handling equipment. The modification incorporates the existing safety features into the new control system. Fuel manipulator malfunctions and accidents that may result are bounded by the analysis performed in support of NRC SER dated June 20, 1979, which conservatively assumes that all the fuel pins in two fuel assemblies are ruptured as a result of a dropped fuel assembly. Therefore, no new accident or malfunction is created by the modification. The manipulator is not taken credit for in mitigating a fuel handling accident and the amount of fuel damage that would occur because of a dropped assembly is not affected by the modification. Therefore, the consequences of an accident, event, or malfunction of equipment is not increased. The seismic capabilities of the manipulator are maintained. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-004-03)

4. MR 95-042*B, (Unit 2), Residual Heat Removal (RHR).

The SE revision clarifies the modification of valves 2RH-704A&B and the routing of the vent line.

Summary of Safety Evaluation: The abandoned steam leak-off line is reconfigured to function as the bonnet vent. It is then routed back to the RHR system at the valve inlets. The joint between the vent and leak-off connection is a compression fitting that is qualified for this application. Since the relieving flow need not be greater than a fraction of a gpm, the vent is constructed of 3/8" tubing that is configured to the desired shape by bending. Approximately one foot upstream of the parent valve, a pipet is welded to the RHR 8" line to connect the vent to a location at a lower pressure than the bonnet. To connect the 3/4" NPD pipet to 3/8" tubing, an adapter is welded to a compression fitting that is qualified for the application. The new routings, as well as their interfaces, are designed, fabricated, and tested to the same standards and criteria as the original systems. The normal configuration is to have the vent valves shut. The shut vent valves render the parent valve unchanged regarding their sealing capability in both directions. When it is required to open a parent valve, the operator first opens its vent valve, located within arm's reach from the parent valve, to relieve the bonnet pressure and then immediately shuts the vent valve. Thus, the parent valves, being located in an area normally accessible with varying radiation doses, are ready for easy opening and the overall radiation exposure is reduced by the modification. Since the parent valves remain shut and need not be manipulated during accident conditions, they are not provided with reach rods.

The reactor is defueled during this installation. This is necessary because the valves modified can not be isolated to allow simultaneous work and RHR system availability. Since the work is performed on systems that handle the primary coolant, precautions associated with such actions, including the control of exclusion of foreign materials from the systems, are strictly enforced. The entire installation is QA-scope and safety-related. A pressure test and initial service leak test is performed as acceptance of the modification. FSAR Figure 9.3-1, "Auxiliary Cooling Systems," is revised to incorporate bonnet vents, valves and tubing added to valves 2RH-704A&B. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-117-01)

5. MR 95-043, (Unit 1), Refueling Cavity Drain System.

The modification upgrades the active and passive draining capability of the refueling cavity drain system. The modification removes the refueling cavity drain barrel, the associated piping to the barrel (including the 4" cross fitting and drop leg), the flapper valve, and the reducing diffuser in the cavity drain. The drain barrel and reducing diffuser are removed because of its ineffectiveness in reducing dose in the area, and the barrel has actually developed into a high dose contributor. The flapper valve and holding mechanism is removed to facilitate the installation of the debris strainer. The new piping includes a 4" tee installed at the location of the existing cross fitting. A new pipe anchor and a rod hanger support is installed downstream of the first isolation valve in the drain line, valve 1WL-01705. A debris strainer is also installed at the inlet of the refueling cavity drain line (in the cavity) to prevent debris from clogging the drain.

Summary of Safety Evaluation: The debris strainer is safety-related/seismic Class I and manufactured from QA materials compatible with the containment and cavity environments. The strainer geometry is designed to minimize the holdup volume of containment spray fluid in the cavity if portions of the strainer do become blocked. Calculations demonstrated that the strainer remains in place under seismic and fluid drag loading conditions. The portions of the installation between the cavity drain opening and normally shut valve 1WL-01705 are QA-scope and safety-related, seismic Class I. Valve 1WL-01705 is procured to provide a QA/safety-related boundary. A calculation showed that leakage past valve 1WL-01705 diverts an insignificant volume of water from the containment sump. A calculation ensures that this valve maintains its integrity during a fuel handling accident. The new pipe anchor provides an analysis boundary for the safety-related, seismic Class I portion of the cavity drain line. The piping and supports in the non-seismic region were evaluated and modified for functionality to ensure the validity of the seismic analysis. The system is leak tested to ensure construction standards for the safety-related piping have been met.

The modification accomplishes the following: 1) Upgrades a portion of the refueling cavity drain line, including the first isolation valve, to safety-related and seismic Class I so that the drain pipe can be relied upon to pass and retain water from the lower refueling cavity; 2) Makes provisions to ensure that the containment spray/ RCS fluid that enters the refueling cavity during a large break LOCA drains and is made available to the containment sump, leaving a minimal holdup volume in the cavity; and, 3) Reduces the dose on containment El. 8' attained during refueling outages by the refueling cavity drain barrel and drain piping. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-074)

Summary of Safety Evaluation: The SE revision incorporates changes to the modification as identified via engineering change request ECR 1999-0181. The ECR documents a change in pipe anchor location from the initial modification. The subsequent requalification of the piping system with the pipe anchor in a new location also requires that an original pipe support remain in place rather than removed as initially stated. This results in a reduction in the number of pipe supports removed by this modification. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-074-01)

6. MR 96-047*A, (Unit 1), Reactor Coolant.

The modification replaces the Unit 1 power-operated relief valve (PORV) block valves (RC-515, T-1 pressurizer RC-431C PORV isolation, and RC-516, T-1 pressurizer RC-430 PORV isolation). The original 3" flex wedge gate valves are replaced with 3" parallel disc gate valves. The installation is coordinated with other U1R25 work when the reactor coolant system (RCS) is vented.

Summary of Safety Evaluation: NRC letter dated January 8, 1998, states our commitment to replace the PORV block valves during the 1998 or 1999 refueling outages. The commitment addresses GL 95-07, "Pressure Locking and Thermal Binding of Safety Related Operated Gate Valves."

Malfunction of the PORV block valves is not considered in the FSAR. The pressurizer safety valves are designed to accommodate the maximum surge rate from a complete loss of load without a reactor trip or other control if the block valves fail shut. FSAR Section 4.3 discusses this design. The block valves isolate failed PORVs. The block valves are remote manual actuated motor-operated valves. The replacement valves reuse the motor operators and have a similar stroke time to the original valves. The new valves meet the original design requirements and correct the design deficiencies of the original valves. The new valves are not susceptible to pressure locking, thermal binding and stem embrittlement and should provide years of reliable service. EPRI TR 103255 discusses a gate valve performance prediction program. The program recommends rounding the disc edges and hard facing disc guides to ensure proper operation under design conditions. The parallel disc design has these design characteristics as applicable and a much lower unseating load than the original valves. These features ensure that the valves perform under design conditions. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-058)

7. MR 96-068*D, (Unit 2), Containment Heating Steam.

The modification removes the containment heating steam system heat exchangers HX-76A-H from both containments during refueling outages. MR 96-068*D disconnects and removes electrical equipment associated with heating steam equipment motors. FSAR Section 1.2 is updated to reflect the heating steam equipment removed from containment.

Summary of Safety Evaluation: The heating steam system is not an accident initiator. Removing aluminum from containment lowers total post-LOCA hydrogen generation, decreasing the possibility of a hydrogen burn inside containment. Therefore, removing the heat exchangers does not increase the probability of occurrence of an accent or event or malfunction of equipment important to safety previously evaluated in the CLB.

The heating steam system is not an accident mitigator. Removing the abandoned systems and reducing aluminum inside containment does not cause a malfunction to equipment important to safety. Therefore, radiological consequences of an accident, event, or malfunction of equipment important to safety remains unchanged.

Reducing aluminum inside containment lowers total post-LOCA hydrogen generation thus ensuring margins of safety associated with the containment design pressure of 60 psi as described in TS 15.4.4, "Containment Tests," are not adversely affected. The modification does not pose a USQ nor does it require a change to the TS. (SE 97-192-01)

8. MR 97-014*C, (Common), 125 V.

The modification transfers the power supply for D-40, G-04 emergency diesel generator (EDG) DC distribution panel, from switch D72-02-03 in panel D-02, to existing spare breaker D-14-18 in panel D-14. Panel D-14 is supplied from D-02.

Summary of Safety Evaluation: G-04 EDG is inoperable during installation when the supply to panel D-40 is deenergized. There is not a required action entry, provided G-03 EDG is operable and aligned to 2A-06. Operations is required to perform the necessary emergency power system lineups. DC control power to 2A-06 is transferred from D-40 to D-28 prior to deenergizing D-40. During transfer of DC control power from normal to alternate, DC control power to 2A-06 is deenergized for a short time (less than a minute). This disables automatic breaker closure and opens schemes for breakers on the bus. This prevents EDG breaker closure in the event of a loss of offsite power. Therefore, a required action for emergency power to 2A-06 is entered in accordance with TS 15.3.7.B.1.f. A required action for 2P-15B safety injection (SI) pump is also entered in accordance with TS 15.3.3.A.2.b during shifting of DC control power to 2A-06. Removal of DC control power to bus 2A-06 does not increase the probability of a fault or other event that could result in the loss of offsite AC power. Operating loads are not affected because of installation controls in place during energized panel work. Relocation of the power supply to panel D-40 and installation of panels D-27 and 1D-202 does not increase the probability of the loss of DC power to plant equipment and does not affect the safety functions of the 125 VDC system.

The modification also installs new safety-related DC distribution panel D-27 and supply power to panel D-27 from switch D72-02-03 in panel D-02, including moving two terminal boxes to make room for panel D-27. One of the terminal boxes contains only spare cables from 1C-129, 2C-129 and D-14. The other terminal box contains circuits for control of G-01 EDG room air dampers. These control circuits are non-safety related. The terminal box and associated conduits and cables are relocated higher on the wall to make room for D-27. G-01 EDG remains operable during terminal box relocation. No loads are transferred to panel D-27 in accordance with this design package.

Panels 1D-202 and D-27 contain fused switches rather than molded-case circuit breakers. The modification corrects problems with age-related degradation of molded-case circuit breakers. Fused switches are less subject to age-related degradation and improves DC system coordination and increases fault clearing capability. New panel D-27 is manufactured for use in QA, safety-related, seismic Class I application and contains UL-listed components. Mounting of the new panel, including conduit rework, is seismically qualified. New cables are Class 1E qualified and meet fire retardency requirements. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-071)

Summary of Safety Evaluation: The new power supply to panel D-40 meets train separation requirements. The addition of D-14 in the supply path to D-40 adds fire Zone 318 (cable spreading room) to the D-40 supply. However, panel D-40 is not required for Appendix R scenarios involving a cable spreading room fire. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-071-01)

9. MR 97-014*D, (Unit 1), 125 V.

The modification replaces 125 VDC circuit breaker panelboards D-04 and D-41 with new fused switch panelboards.

Summary of Safety Evaluation: Inverters 2DY-04 and DY-0D are temporarily supplied from swing safety-related panel D-302 during replacement of D-04. This allows the Unit 1 and 2 Yellow instrument buses to remain energized during replacement of panel D-04.

The modification replaces panel D-21 and transfers the power supply to panel D-21 from D-13 to D-27. It transfers Unit 1 Train B loads from panels D-13, D-14, D-18 and D-19 to panels D-21 and D-27. The modification also transfers Unit 1 Train A loads from panel D-12 and D-22 to panels D-11 and D-16. It energizes panel 1D-202 (installed per MR 97-014*C), and transfers Unit 1 non-safety related loads to panel 1D-202. The modification also installs a 20 A DC test receptacle in the maintenance electrical shop and supplies power to the receptacle from 1D-202.

Several loads supplied from the 125 VDC system are deenergized during the installation. Since Unit 1 is in a refueling outage, most accidents for which the affected loads are required to be operable are not possible. DC control power to the affected buses is deenergized for a short time during shifting of DC control power supplies to 13.8 kV, 4.16 kV, and 480 V switchgear. This does not increase the probability of a fault or other events resulting in the loss of offsite AC power. No safety-related loads are added, removed, or modified within the scope of this modification. Only the DC power supplies for the affected loads are reconfigured.

New safety-related panels are manufactured for use in QA, safety-related, seismic Class I applications. The modification corrects problems with age-related degradation of molded-case circuit breakers. Fused switches are less subject to age-related degradation and improve DC system coordination and increase fault clearing capability. The new panels contain UL-listed components. New cables meet fire retardency requirements. Mounting of panels and raceways is seismically qualified. The new panels have ratings equivalent to those of the original panels. This modification does not introduce new failure modes or reduce the independence of the main DC buses, including interim conditions. Accidents or events associated with the loss of a single DC train were previously evaluated. The only failure modes associated with the new DC panels and associated cabling are open or short circuits. These failure modes exist with the original equipment and have been previously evaluated. Original safety-related panel D-302 temporary becomes D-04 during replacement of panel D-04. This meets TS requirements for having all four safety-related DC buses in service. This modification does not affect the degree of independence of safety-related DC system trains. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-077)

Summary of Safety Evaluation: DC control power to 480 V bus B-04 is transferred from panel D-13 to D-41 via MR 96-022. The design and installation of that work was evaluated via SE 96-072. Final connections, post-maintenance testing and transfer of DC control power from D-13 to D-41 are procedurally controlled to provide better configuration control. During transfer of DC control power from normal to alternate, DC control power to bus 1B-04 is deenergized for a short time. This disables automatic stripping of non-safety related loads and starting of safeguards loads in the event of the loss of offsite power and safeguards sequencing. During Unit 1 cold or refueling shutdown, the only safeguards loads of concern are P-032C service water (SW) pump and 1P-010B residual heat removal (RHR) pump. P-032C is inoperable during transfer of DC control power to 1B-04 and a SW action statement is entered. Since only a single train of RHR is not required. 1P-015B safety injection (SI) pump is inoperable during transfer of DC control power to 1B-04 to compensate for an increase in emergency diesel generator (EDG) loading because of the absence of bus stripping in the event of a loss of offsite power. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-077-01)

10. MR 97-066, (Unit 2), Safety Injection (SI).

The modification corrects a deficiency in the design for SI-887 SI test line relief valve. The relief valve protects the SI piping from RCS back leakage through the injection line check valves. The original sizing of the relief valve was based upon check valve leakage criteria of 2 cc/hr/in, resulting in a required relief capacity of 1 gpm with a lift setpoint of 1745 psig.

Summary of Safety Evaluation: The modification increased the relief valve relieving capacity and increases its temperature rating. The modification installs a new nozzle assembly and changes the spring material from carbon steel to Inconel. The setpoint remains unchanged. The valve is periodically tested via the relief valve test program. There is no increase in the probability of the relief valve sticking open, or failing to relieve. Loss of reactor coolant is the only evaluated accident that this relief valve could affect. However, the RCS leakage is limited by the back leakage through the injection line check valves. Leakage would be collected in the pressure relief tank (PRT) and contained in the waste disposal system. Leakage of approximately 12 gpm is well within the capacity of one charging pump, and makeup is available even under the loss of offsite power condition. Therefore, the change can not initiate a loss of reactor coolant accident.

Blowdown from the relief valve is directed to the PRT. The PRT is not adversely affected by the small increase in discharge capacity of SI-887. The PRT is protected from overpressure by a rupture disc and is equipped with level indication. Likewise, the slight increase in relief capacity does not adversely affect the SI system ability to perform its safety-related function. Leakage through the relief valve during SI system operation would essentially reduce system capacity. However, Calculation N-95-0148 concluded that a diversion of 7 lbs/sec, or 50 gpm, or high head SI water from a single operating train at 1400 psig does not prevent the SI system from performing its design function. Therefore, the use of an approximately 12 gpm relief does not challenge the operability of the SI system.

The relief valve is removed and rebuilt while its respective unit is in a refueling shutdown, when the SI system is not required to be operable per TS 15.3.3 and 15.3.15. The modification does not pose a USQ nor does it require a change to the TS. (SE 97-167)

11. MR 97-068*B/D, (Unit 1 and 2), CVCS.

MR 97-068*B and D install independent nitrogen backup systems for each unit on each units turbine hall El. 8'. The nitrogen systems are connected to the nitrogen bottles to the 1&2RK-24 and the vari-drive actuators to provide an alternate pneumatic supply to the charging pump speed controls.

Summary of Safety Evaluation: Components are constructed of materials capable of meeting or exceeding the instrument air (IA) system pressure and temperature ratings. Associated system components remain idle (with the exception of system testing or necessary operation) and are manually actuated when needed. Unit 1 tie-ins at the vari-drive actuators are installed via MR 97-068*B by removing a single charging pump from service at a time. Upon completion of each unit modification, valves are aligned to perform full functional testing of the nitrogen system with the units at power.

Nitrogen gas is not flammable and does not increase the probability of an Appendix R event. Probability of uncontrolled dilution as described in FSAR Chapter 14.1.4 is unchanged since equipment malfunction is not affected. Bottles are secured to the turbine hall wall and are handled in accordance with approved procedures, including ruptured disks to ensure low missile probability. PAB habitability or equipment accessibility because of air quality is not adversely affected since controller system consumption rates are of an insignificant magnitude with respect to natural convection resulting from nearby equipment. The RK-24 controllers and a single P-2 actuator are aligned to operate on nitrogen, using pressure slightly above normal IA operation, but within IA design pressure limits. The controller and actuator response remain unchanged because the charging pump speed control components receive a regulated pressure supply and are designed to operate using air or dry gas. A loss of charging pump speed control is not considered credible because the IA remains aligned to the RK-24 controllers at all times. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-025)

12. MR 97-081*A, (Common), Service Water (SW).

The modification adds a gearbox and motor operator to manual valves SW-652 and SW-653 so that future modifications can be performed to add safety injection (SI) logic to the control scheme of each valve.

Summary of Safety Evaluation: The modification consists of modifying manual service water valves SW-652 and SW-653, on the inlet side of spent fuel pool (SFP) heat exchangers HX-13A&B, into motor-operated valves, similar to the valves SW-2930A and SW-2930B, on the outlet side of the heat exchangers. The new valves (relabled to SW-2927A and SW-2927B) are powered from motor control centers (MCCs) 2B-42 (spare breaker 7J) and 2B-32 (spare breaker 8F), respectively. In addition, each valve has a local control station installed with indicating lights for valves position. Adding SI logic to the control scheme of new valves SW-2927A and SW-2927B provides redundancy with original valves SW-2930A and SW-2930B that allow SW to be positively isolated from SFP heat exchangers HX-13A&B and diverted to a more critical area following a SI signal from either unit. Installation and testing is planned accordingly to ensure that one SFP heat exchanger remains in service to provide decay heat removal to the SFP.

No new changes are made that could cause failures of the SW system and its ability to cool the SFP. The SW system is assumed available during accidents or events described in the CLB. In addition, neither the SW system nor its components are identified as initiators of accidents or events as described in the CLB. Calculations assessed the affects of adding the new MOVs to various plant systems. The calculations reflect that the SW piping at the SFP heat exchangers are unaffected by the additional weight of the new operators. MCC and EDG loading are unaffected because the calculations do not take into account intermittent loads such as MOVs. Calculation has shown that the new MOVs have adequate stem thrust and torque during worst case undervoltage conditions, and the new MOVs still perform their safety-related function during hot-smart-short overvoltage conditions. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-024)

13. MR 97-081*B, (Unit 2), Service Water (SW).

The modification enhances the ability of the SW system to provide necessary cooling to essential loads during a design basis accident. It provides for redundant isolation of SW to the SFP heat exchangers. The modification provides an auto close signal to the SW inlet valves to the SFP pool heat exchangers upon an SI signal from either unit independent of the number of SW pumps that are started. Also, remote control switches and indicating lights are installed on C-01 in the control room, for control of the SW inlet valves to the SFP heat exchangers. Existing spare cable and spare conductors from in-service cables are used for electrical connections between the motor control center and C-01, and between C-01 and the safeguards racks in the cable spreading room.

Summary of Safety Evaluation: Testing involves cycling the valve with the new control switch and verification that the SI signal causes the valve to shut automatically by temporarily jumpering terminal points in the safeguards relay racks. The Unit 1 SI contacts were tested via PBTP-099 and PBTP-100 that are evaluated separately. The Unit 2 safeguards relay contacts were wired out to terminal strips and tested per MR 97-081*C. Repositioned control switches for SW header valves SW-2870 and SW-2869 are also tested to ensure proper operation. Short entry into the applicable TS required action statement occurs for this portion of the testing.

The current SW flow analysis shows that there is little flow margin for operability of the containment fan coolers (CFCs) when the lake water temperature approaches its design limit. Providing the redundant isolation of the SFP heat exchangers ensures that more water is available for the essential loads. Isolating flow to the SFP heat exchangers during an accident does not create a problem with SFP cooling. Adequate time is available to either reestablish flow to the SFP heat exchangers or provide alternate means for cooling. Plant design ensured SW to the SFP heat exchanger is isolated during an SI if less than four SW pumps start. The design also considered such issues as system overpressurization, SFP heat exchanger shell side vacuum (with only the inlet valves shutting) and water hammer. These issues were evaluated and determined that the SW system integrity is unchanged.

The equipment affected by the modification is not an initiator of accidents or events. The modification enhances the ability of the SW system to provide cooling to essential loads during an accident. Based on this, the design change does not increase the probability of accidents or events and does not increase the radiological consequences of accidents or events. Procedural controls used during installation ensure that the installation and testing activities do not affect equipment important to safety. One train of SFP cooling remain available at all times during installation and testing. The SW flow model is met by not allowing SW to be aligned to more than one SFP heat exchanger at a time. The basis for the SW system operational requirements described in TS 15.3.3-D is that the system is capable of providing cooling water to various essential plant equipment. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-053)

Summary of Safety Evaluation: The SE revision addresses an evaluation of the interim condition where SW-2870 and SW-2869 are deenergized at different times to allow relocation of the control switches on control panel C-01. Power to SW-2870 and SW-2869 is removed from service (not at the same time) to allow for relocation of the control switches, but the valves remain open to allow the SW ring header continuous flow path to remain in service. The valves may still be manually operated as necessary to isolate the main loop headers in the event of a piping failure such that the safe shutdown requirements of the SW system are not impacted. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-053-01)

14. MR 97-109, (Common), 125 V.

The modification installs a switch for each safety-related battery charger contactor that isolates the external control cables and directly energizes the contactor closing coil to close the contactor. The switches have two positions: "Auto" which does not affect contactor operation, and "Close," which isolates remote control circuits and interlocks and closes the contactors. The switches can be locked. All switches accept the same key. The switches are mounted on the contactor enclosure doors.

Summary of Safety Evaluation: The modification does not affect redundancy or separation of the battery charger contactor control circuits. It does affect the contactors for the safety-related battery chargers. However, these contactors are not associated with initiating events or cause any of the accidents or events previously evaluated in the CLB. The switches are purchased for use in safety-related applications in accordance with QA program requirements. Installation of the switches is in accordance with installation work plans. The switches are tested following installation to ensure that contactor functions are operational. The new switches are supplied with key locks to prevent operation by unauthorized personnel. Operation of the switches is procedurally controlled to also prevent improper use. The switch type is similar (type and manufacturer) to the original remote and contactor control circuits. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-063)

Summary of Safety Evaluation: The following contactors have switches installed: 1B42-3212H, 1B42-391, 1B42-491, 1B42-494, 2B42-391, 2B42-394, 2B42-4212B, 2B42-491. References to the system operating procedures are included in the procedures that address abnormal contactor operation, including Abnormal Operating Procedures (AOPs) and Alarm Response Procedures (ARPs). The procedure changes improve the timeliness of battery charger restoration following an Appendix R fire or other event that damages the normal contactor control circuits. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-063-01)

15. MR 97-113, (Unit 2), Reactor Coolant System (RCS).

The modification occurs when the reactor is defueled. The SE revision addresses a refueling outage schedule that requires the implementation of the modification to be extended beyond the defueled time. Installation can occur up to the time when the start of the refueling cavity drain activity occurs. The modification installs six relief valves and associated inlet and outlet piping/tubing. The valves are to relieve a potential thermal overpressure that could occur in the RCS isolated piping sections between normally closed valves inside containment following a LOCA or a main steam line break (MSLB).

Summary of Safety Evaluation: The sections of piping modified are in the reactor vessel level indication system (RVLIS) and the RCS. The RCS piping sections are normally isolated when the reactor is operating at greater than cold shutdown conditions. The safety relief valves, as indicated in Calculation M-09334-358-RC.1, are selected to relieve excess fluid so that the subject piping and associated components are adequately protected from overpressure due to external heating during a LOCA or MSLB.

The RCS is depressurized during installation in Unit 2 containment because the reactor is in cold shutdown and the reactor is being loaded during U2R23. This condition allows isolation of the locations affected by the modification by one isolation valve only. It is assumed that these valves are designed for much higher than water column static pressure and seismically qualified. This is adequate assurance for maintaining an RCS pressure boundary during refueling coincident with a seismic event. The work is performed on systems that handle the primary coolant; therefore, precautions associated with such actions, including the control of exclusion foreign materials from the systems, are strictly enforced. The entire installation is QA-scope and safety-related. A pressure test and an initial service leak test are performed as acceptance for the modification.

The valves and pertinent piping and tubing installed are properly qualified and installed in accordance with approved codes and standards, including the original piping specifications. A seismic review determined the design meets appropriate seismic requirements. The structural integrity of the isolated piping is maintained by the modification and no other equipment is affected. The modification does not affect containment integrity. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-071-01)

16. MR 97-134, (Unit 2), CVCS.

The modification installs two relief valves upstream of 2CV-285 and 2CV-386 and associated inlet and outlet piping/tubing. This alleviates an overpressurization condition that could occur in isolated piping sections inside containment during accidents like a LOCA and main steam line break (MSLB) inside containment.

Summary of Safety Evaluation: The chemical and volume control system (CVCS) piping upstream of valves 2CV-285 (between valves 2CV-1299 and 2CV-285) and 2CV-386 (between valves 2CV-386 and 2CV-317A&B) is modified. The newly installed relief valves are in fact bypass valves to the parent valves that open on high pressure in the subject isolated piping sections and relieve some liquid downstream into the lower pressure regions. Valve connections are made in 3/4" main pipe by use of a socket welded Tee fittings with a pipe-to-tube adapter welded in the branch side to accommodate the dimensional difference. The bypass is constructed of 1/2" tubing configured to the desired shape by bending. Joints within the tubing portion of the bypasses (including the relief valves) are compression fittings qualified for the application. The safety relief valves can relieve excess trapped water so that the subject piping and associated components are adequately protected from overpressurization because of external heating during a LOCA or MSLB.

The new relief valves are installed in a dead leg that does not change the CVCS flow configuration. The valves relieve a small amount of contaminated water back to the parent system when operated. Since no new flow paths are created for uncontrolled loss of reactor coolant and the operability of the parent systems is maintained, the modification does not alter system or component functions. Each relief valve, if stuck in the open position, is capable of discharging 0.3 gpm. Loss of RCS inventory occurs if CVCS is in an unusual valve lineup that induces full RCS pressure across it. In that case, the valve is capable to release 0.3 gpm of water. Leakage of 0.3 gpm is readily detectable by instrumentation, water inventory balance, or direct observation. Once the undesired leak is detected, the operator follows TS 15.3.1.D that prescribes to either eliminate the problem, or start a reactor shutdown. Installation occurs with Unit 2 in cold shutdown, and defueled; therefore, no operational limitations are necessary during the work. However, the work is performed on systems that handle the primary coolant. Therefore, precautions, including control of foreign material exclusion, is strictly enforced. The installation is QA-scope and safety-related. A pressure test and an initial service leak test is performed.

The valves and pertinent piping and tubing are properly qualified and are installed in accordance with approved codes and standards, including the original piping specifications. A seismic review revealed that the design meets appropriate seismic requirements. Since the parent piping contains water moving at a velocity of 11 fps (a moderate velocity), analysis of the tubing in regard to the flow induced vibrations was not performed. The structural integrity of the isolated piping is maintained and no other equipment is affected. Since the modification does not affect containment integrity, there is no increase in the consequences of an accident. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-118-01)

17. MR 98-002, (Common), Plant Process Computer System (PPCS).

The SE addresses a required change to the Basis of TS 15.3.5 because of MR 98-002. The new plant process computer seismic safety parameter display system (SSPDS) replaces plasma displays in Unit 1 and 2 auxiliary safety instrumentation panels (ASIPs) via this modification.

Summary of Safety Evaluation: The plasma displays located in the ASIPs are a part of the safety assessment system (SAS) that must be qualified to operate during and after a design basis seismic event. When the displays were installed in the mid-1980's, plasma displays were the best choice to satisfy seismic qualification. Other types of seismically rugged display types are now available. There is no technical or regulatory justification for maintaining the plasma displays.

The modification removes the plasma displays; therefore, the word "plasma" is to be deleted from the following sentence in TS 15.3.5: "A second backup display of subcooling information is available on seismically qualified plasma displays which receive input..." The TS section describes the main subcooling margin displays and the primary, secondary, and tertiary means of determining the subcooling margin. The term "plasma" is merely descriptive of the type of display currently in place and is not a basis for the margin of safety for the subcooling indication. The modification nor its change to TS poses a USQ. (SE 99-052)

18. MR 98-002, (Common), Safety Assessment System (SAS)/Plant Process Computer System (PPCS).

    The work orders (WOs 9907892, 9907894, 9907895 and 9907893) disconnect the data link between the ISFSI temperature monitoring system and the PPCS. The interim conditions created affect access to data normally used to assist the operator and technical support personnel in the surveillance of cask thermal performance. The Certificate of Compliance (C of C) requires daily monitoring of the temperature but does not specify how the monitoring is accomplished. The FSAR notes that continuous temperature monitoring is provided through PPCS.

    Summary of Safety Evaluation: The PPCS (including data links between the RMSCT2) provides supplementary information to the operator and technical support personnel to assist in the normal operation of the steam supply system. It also is used as an alert to off-normal conditions. Adequate instrumentation exists for the operator to operate the plant in a safe manner, regardless of the availability of the PPCS. The PPCS and its associated display are not identified as an initiator of accidents identified in the CLB. The SAS/PPCS subsystem (SSPDS) provides the primary display location for the core exit temperature. The SSPDS functions of the multiplexers are not affected by the modification. The inputs in multiplexer cabinet C-178 are not handled during the modification. The temporary laptop personal computer (PC) can not propagate to permanent plant equipment. The "A" PPCS computer and RMS CT1 are confirmed operable prior to the modification. Temporary disconnects of data links exist for a short duration (less than one shift for each data link). However, if for some reason, connection of the temporary laptop PC to the multiplexer causes problems with the application controlling data to the "B" PPCS, data flow to the "A" computer and data flow between the multiplexers is not affected. The data link connection can easily be restored by personnel monitoring the data link. Contingency plans exist in approved plant procedures. The modification does not pose a USQ nor does it require a change to the TS.

    10 CFR 72.48 Evaluation Summary: The ISFSI temperature monitoring inputs to the PPCS provide for monitoring the differential temperature between the inlet and outlet of cask versus an established maximum allowable differential temperature. The PPCS and its associated display devices are not identified as an initiator of accidents or off-normal events identified in the ISFSI licensing basis. The temporary laptop PC is electrically isolated through original fiber optic cable from the ISFSI temperature monitoring system so that an electrical fault in the laptop PC can not propagate to permanent plant equipment. Existing approved procedures specify that local surveillance of the casks at the ISFSI is required when the PPCS is unavailable. The local temperature monitoring activities are performed in conjunction with daily visual inspections of the cask air inlets and outlet for screen blockage or degradation. The temperature monitoring can be performed in the ISFSI gatehouse (disconnecting the ISFSI input to PPCS does not place the ISFSI gatehouse jack panel out of service). The modification does not pose a USQ, significant increases in occupational exposure, significant unreviewed environmental impact, nor changes in the license conditions of the C of C are created. (SE 99-064)

19. MR 98-002*B, (Common), SAS and PPCS.

    The modification installs new plant computer equipment in the computer room; installs network components and two temporary workstations (consisting of one CPU, monitor, keyboard and mouse) in the control room; installs cables to jumper the parallel computer points on terminal strips in the original multiplexer cabinets to barrier terminal strips in new controller cabinets; installs fiber optic cable between the computer room and the control room; and installs fiber optic cable between the computer room and the technical support center (TSC). The new system is set-up to allow the future connection of serial data (as controlled through the site acceptance test) from the computer in the multiplexer cabinets, radiation monitoring system (RMS), the ISFSI temperature monitoring system and the secondary performance monitoring system recorder to support a site acceptance test.

The new computer components are not powered up during this portion of the modification.

Summary of Safety Evaluation: The PPCS and associated display devices are not identified as an initiator of accidents identified in the CLB. The SSPDS functions of the multiplexers (primary display of the core exit temperature data) are not affected by the installation of cabling between the multiplexers and the new I/O controller cabinets since the core exit temperature inputs are handled in a different portion of the multiplexer cabinets. To ensure against inadvertent termination of the wires in the multiplexer and I/O controller cabinets, the installation work plan includes additional verification of terminations. Parallel connection to the parallel computer point inputs in the original multiplexer cabinets monitor the original non-safety related contact inputs that provide signal isolation between the PPCS computer and the associated instrument loop. The new PPCS is not connected to the computer in the multiplexer as part of this modification. As such, the data flow to the "A" or "B" PPCS computer and data flow between the multiplexers is not affected by the modification and SAS/PPCS is 100% available. Temporary power for the new computer server, controller cabinets and workstations is provided from non-vital buses to control the loading on vital buses.

Loading on the buses was evaluated to ensure that the new PPCS components do not affect the vital instrument buses. Interim equipment locations were evaluated for potential Seismic II/I concerns to ensure that non-safety related components do not affect safety-related components. Temporary equipment is installed in locations well separated from safety-related equipment or tie-downs are provided to prevent non-safety related equipment from impacting safety-related equipment. The parallel operation of new and original computer components (once the new PPCS is powered up during the site acceptance test) results in higher heat loads on the computer room/control room HVAC system. The heat loads remain below the capacity of the computer room/control room HVAC system during the period when the new equipment is installed. The number of core exit temperature measurement points is not affected by the modification. Existing approved procedures define the methods for implementing TS monitoring and surveillance when the SAS/PPCS system is not available. No other TS related components are affected by the modification. The new PPCS is not connected to the computer in the multiplexer as part of the modification. As such, the data flow to the "A" or "B" PPCS computers and data flow between the multiplexers is not impacted by the modification and SAS/PPCS is 100% available. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-083)

20. MR 98-013, (Common), Potable Water.

The modification installs water treatment equipment to improve the well water drinking water quality.

Summary of Safety Evaluation: The SE determines the acceptability of routing the new water treatment reject water to the PBNP effluent sump. It also addresses a change to FSAR Appendix I, Section 1.3 to denote this configuration. The change to the FSAR requires the addition of the sentence, "Potable water system reverse osmosis unit concentrate discharge and filter backwash are drained into the effluent sump". The new water treatment uses a reverse osmosis (RO) system that requires a discharge at a maximum of 13,000 gpd. This discharge is routed via a new underground pipe and gravity drained to the PBNP effluent sump located near the sewage treatment plant. Effluent drained to the sump is pumped to the retention pond by the use of two existing pumps, P-109A&B. The added flow to the retention pond because of the RO system discharge is minimal such that the settling capacity is maintained within WPDES limits. The effluent of the retention pond is routed by existing piping to the plant circulating water discharge. During and after construction, neither the pumps nor existing effluent discharge piping require modification, disruption, or removal from service.

The addition of the potable water treatment discharges to the effluent sump does not introduce factors that could increase the probability of an accident evaluated in the FSAR. The functionality of the effluent sump remains the same. The activity adds potable water system rejected water to the plant effluent sump. The additional water is not radiological; therefore, the activity does not increase the radiological consequences of an accident, event or malfunction of equipment important to safety. Since the existing piping and pumping equipment used to discharge the effluent sump remains the same, the possibility of an accident of a different type is not created and does not increase the probability of equipment malfunction important to safety. No margins of safety are reduced by the addition of the potable water effluent discharge. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-033)

21. MR 98-013, (Common), Potable Water.

The modification installs a 500 gallon liquefied petroleum (LP) gas tank approximately 10' west of the potable water wellhouse. The tank is to be used to supply LP gas to the new gas heating system in the potable water wellhouse. The tank is to supply the LP gas via an underground pipe.

Summary of Safety Evaluation: The installation is in accordance with standards contained in the National Fire Protection Association (NFPA). The addition of the LP gas tank does not introduce factors that could increase the probability of occurrence of an accident evaluated in the CLB. No safety-related equipment or piping is degraded or modified by the installation, therefore, the probability of an accident is not increased. The LP gas tank is outside the protected area fence and has no affect on the fire protection manual description of accidents or events. Additionally, the new tank is located more than 50' from the protected area and has no impact on equipment important to safety or control room habitability as evaluated in the CLB. For these same reasons, the tank does not influence the possibility of malfunction of equipment important to safety not evaluated in the CLB. The installation of the LP gas tank is not in a radiological area and has no impact on the radiological consequences of an accident and the possibility of an accident or event not previously evaluated in the CLB is not impacted. For added protection of the LP gas tank, jersey barriers are placed in front of the tank for prevention of vehicular contact. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-093)

22. MR 98-024*U, (Unit 1), Service Water.

MR 98-24*U revises the original Unit 1 Train A and B safety injection (SI) SW isolation logic such that the SW isolation valves shut on Unit 1 SI signal within their associated train. This consists of abandoning in place the original Unit 1 4-out-of-6 SW pump logic that isolates SW from non-essential loads if less than 4 SW pumps start following and SI signal. The logic change affects the following valves: SW-2816 (inlet to PAB coolers), SW-2817 (inlet to water treatment area coolers), SW-2930A (outlet from SFP HX-13A), SW-2930B (outlet from SFP HX-13B), SW-LW-61 (inlet to radiation waste coolers) and SW-LW-62 (outlet from radiation waste coolers), and 1SW-2880 (inlet to Unit 1 turbine hall coolers). The SI signal to shut 1SW-2880 is removed. New motor-operated valves SW-2927A (inlet to SFP HX-13A) and SW-2927B (inlet to SFP HX-13B) had SI logic installed such that they isolate on SI signals. The Unit 1 SI logic for SW-2927A&B is revised so that the SI signal comes from the same relays (1-SW-AX for SW-2927B and 1-SW-BX, for SW-2927A) as the original valves listed above. In addition, spare contacts from these same relays are wired out to terminal strips and tested for future use on valves SW-527 (to become MOV SW-4478, inlet to WT area coolers) and SW-502 (to become MOV SW-4479, inlet to PAB coolers).

Summary of Safety Evaluation: The modification improves the SW margin to essential loads, (e.g., the CFCs during design basis accident conditions). The most limiting design basis accident is with 5 operable SW pumps. Presently, under this scenario, SW to non-essential loads is not isolated. Removing the 4-out-of-6 SW isolation logic allows SW to be isolated to these loads on SI signals and improves margin in the system. SW isolation capability is removed from 1SW-2880 to allow the turbine to be shutdown and secured in a safe manner, (e.g., cooling remains to the turbine lube oil coolers). Adding SI logic to new valves SW-2927A&B, SW-527 (SW-4478) and SW-502 (SW-4479) also increases SW margin when the SW Upgrade Project is complete because these valves are redundant to SW-2930A&B, SW-2817 and SW-2816, respectively, in power supply and SI signal. The redundancy ensures that SW to non-essential loads is isolated even under a non-required actions, SI, single train failure scenario. Unit 1 is in cold or refueling shutdown with no refueling operations in progress. Unit 2 may be in any mode of operation during installation and testing of the modification. The installation and testing only takes place with one train of safeguards and valves deenergized at a time. Voluntary action statement entries for standby emergency power to 1A-05/B-03 automatic SW isolation valves inoperable, and motor-driven auxiliary feedwater pumps inoperable are entered in accordance with appropriate TS sections. Procedure provisions ensure opposite train equipment remains available during these times. Testing involves stroking non-essential SW isolation valves to functionally verify the SI scheme for each train prior to starting work on the other train. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-080)

23. MRs 98-037*A, 98-037*B, (Unit 1 and 2), Safety Injection (SI).

The modification removes SI-885A&B and replaces the components with a section of 3/4" pipe to provide pressure locking protection. Additionally, it installs normally shut vent valves on the bypass line to improve filling, venting, or draining activities. The SI-885A&B check valves were initially installed as a means of pressure lock prevention for double disk gate valves SI-852A&B. The original configuration vents excessive bonnet pressure to the header downstream of SI-852 via 3/4" pipe and SI-885 check valves. To ensure the SI-852 valves are capable of opening and meeting timing requirements, proper operation of the SI-885 valves is required. ASME Section XI-IWV, 1986 edition, requires check valves with a safety-related function be tested or actuated quarterly. The original configuration did not provide a practical means of performing functional testing of SI-885.

Summary of Safety Evaluation: The modification does not increase accident probability as described in the CLB. The potential to create new accidents is not credible. The modification only involves routine activities (e.g., cutting, grinding, welding) that are controlled by approved plant procedures. The modified piping is in accordance with applicable construction codes, per seismic Class I requirements. Initial service leak testing is performed as permitted by the installation code, using routine alignments of plant equipment. The new valves installed are normally shut, and procedurally controlled to ensure coolant inventory is not lost upon SI-852 opening and system actuation.

The replacement run of pipe without obstructions serves as the new and more reliable means of SI-852 bonnet pressure relief. The modified system includes only passive pressure boundary components meeting design temperature and pressure requirements of the system that introduce no new malfunction modes. The new vent valves were manufactured to ASME Section III standards (minus the N-stamp) which includes a liquid penetrant examination and hydrostatic test to full design pressure. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-176)

24. MR 98-054, (Unit 1), Emergency Diesel Generators (EDG).

The modification replaces the 165°F fusible links of the G-01 and G-02 EDG exhaust fan discharge fire dampers, with 286°F fusible links. The higher temperature fusible links prevent inadvertent closure of the G-01 and G-02 EDG exhaust fan discharge fire dampers. The fire dampers prevent inadvertent isolation of the G-01 and G-02 EDG exhaust fans, which would compromise operability of the EDG engines. The use of the higher temperature fusible links precludes the melting and subsequent closure of these fusible links under high energy line break (HELB) scenarios, which do not result in turbine building temperatures significantly above 200°F. The temperatures in the turbine building are limited to approximately 200°F because of main steam isolation valve (MSIV) closure. The EDG room exhaust fans, W-12A&B, would thereby remain available to provide the ventilation necessary to support G-01and G-02 EDG operability.

Summary of Safety Evaluation: The melting of these fusible links, to permit fire damper closure, is not a nuclear safety function of the affected fire dampers. The increase in fusible links temperature does not result in degradation of the fire barrier characteristics (1.5 hr rating/Class B) as reflected in FPER Section 5.6 for Fire Zones 308 and 309. Since the new fusible link retains the required U.L. Class B requirements, the fire damper continues to be sufficient to provide the requisite Appendix R protection from the hazard poses in the condensate storage tank area. The replacement fusible links are rated at a higher force than the original fusible links and therefore provides greater assurance against closure during a seismic event. In addition, since both the original and replacement fusible links are U.L. Class B qualified, the probability of component failure is not increased.

During installation, the impairment of the associated fire rated barriers requires surveillance of the fire zone in accordance with the requirements of FPER Section 5.6. Post-installation testing of the affected fire dampers verifies the operability of the fire damper. The addition of the higher temperature fusible links allows the dampers to remain open to maintain complete operability of the associated EDGs during a HELB. Therefore, the modification decreases the probability of an equipment malfunction during a HELB. The modification is associated with TS 15.3.7 as it relates to EDG ventilation system operability and the resulting relationship with

EDG operability. Addition of higher temperature fusible links nominally increases the margin of safety by extending operability of the associated fire dampers during a postulated turbine building HELB. This in turn extends operability of the G-01 and G-02 EDG room exhaust systems and ultimately EDG operability. The possibility of EDG derating as a result of higher combustion (intake) air temperature is reduced. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-060)

25. MR 98-075, (Unit 2), RCS.

Baffle-former bolts connect the baffle plates to the formers in the lower internals portion of the reactor vessel. The new baffle-former bolting design replaced a predetermined set of the bolts in a selected pattern. Some of the original baffle-former bolts remain if they were located in "non critical" bolt locations, and one bolt in a "non critical" location was removed and not replaced. The bolting material as well as the bolt design is changed by the modification. The SE revision addresses the design change associated with the newly installed bolting pattern and the new bolt design for Unit 2 with RCS temperature <350°F. It is not intended to encompass the means and methods of the inspection and actual bolt replacement as these issues were addressed separately.

Summary of Safety Evaluation: The baffle-former bolt design allows for leaving bolts in place with indication in non-critical locations or removing bolts in non-critical locations and not replacing them. Baffle-former bolts are not considered accident initiators. The new baffle former bolt acceptable bolting pattern is determined, through detailed analysis as presented in the PBNP specific WCAP-15133, that maintains the functionality of the lower internals. Based on information contained in WCAP-15133 and operational restrictions the modification does not increase the probability or occurrence of an accident or event previously evaluated in the CLB, create new failure modes or new accident initiators, increase the radiological consequences of an accident, event or malfunction of equipment important to safety or decrease margins of safety. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-112-01)

Summary of Safety Evaluation: The SE revision only adds reference to ECR 99-0036 and Westinghouse letter NSD-E-MSI-99-036 based on the actual work that was performed on Unit 2. No changes to the prior evaluations exist. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-112-02)

26. MR 98-093*A, (Unit 1), Containment Penetrations.

The modification removes the nitrogen purge valves from the containment side of the following containment electrical penetration assemblies (EPAs): Q-3, Q-4, Q-5, Q-6, Q-7, Q-8, Q-9, Q-10, Q-11, Q-12, Q-13, Q-14, Q-16, Q-17, Q-18, Q-19, Q-23, Q-24, Q-25, Q-26, Q-27, Q-38, Q-40, Q-42, Q-47, Q-49, Q-51, Q-53, Q-55, Q-56 and Q-57. The purge valves functioned as containment isolation valves (CIVs) in their original configuration; however, they do not satisfy requirements for containment isolation.

MR 98-093*A removes the purge valves and associated tubing from the monitor ring of the EPAs and installs a welded pipe cap over the hole on the monitor ring left by the removed valve and tubing. The cap is fabricated from SA-105 carbon steel. Each affected EPA is leak-checked upon completion of the installation of the welded cap. The new pipe caps and the areas of the EPA monitor rings where paint is to be removed to facilitate installation of the cap are painted upon completion of construction.

Summary of Safety Evaluation: Equipment important to safety is not adversely affected by the modification because the pipe cap is qualified by DE&S Calculation No. 872907.C04-01, and meets the requirements of ASME Section XI, Subsection IWE and the original design requirements of ASME Section III, 1965. The pressure and seismic integrity of the EPAs is not adversely affected by the modification. Removing the valves and installing the pipe caps improves the pressure integrity of the EPAs. The processes used to remove the original valves and install new pipe caps do not damage the conductors in the EPAs. The EPAs are purged with dry nitrogen to remove moisture from its internal components such that the manufacturer's dryness requirements are maintained. The control room is notified prior to welding operations so that operators are aware of potential false data and alarms. Equipment that could be actuated or damaged as a result of the false signals are secured.

The paint for the carbon steel pipe caps is selected and applied in accordance with applicable PBNP procedures and standards for painting in containment. Demolition, installation and post-modification testing is performed to applicable procedures and standards.

The containment system is used for accident mitigation. The modification does not adversely affect the containment pressure boundary; therefore, there is no increase in the probability of failure of this system. Electrical and instrumentation functions remain unaffected. Removal of the valves and installation of the pipe caps ensures no adverse affect on containment integrity since the possibility of a leaking valve or a valve being left open is eliminated. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-067)

27. MR 98-112, (Unit 1), Reactor Internals Lifting Rig.

The modification installs a replacement stainless steel platform on the Unit 1 reactor internals lifting rig. The original platform had an unqualified coating that required its removal from containment. The replacement platform does not require coating.

Summary of Safety Evaluation: The new platform was fabricated. The replacement platform maintains the requirement for providing a safe working surface for maintenance personnel to attach the lifting rig to the reactor internals. Installation of the new platform occurs when the lifting rig is positioned on containment El. 66'. Unit 1 is in cold or refueling shutdown during installation. No work is performed while the lifting rig is in or over the reactor cavity. The replacement platform does not have an adverse impact on the reactor internals lifting rig. The previously established design margins required by NUREG 0612 "Control of Heavy Loads," and ANSI N14.6-1978 are not reduced. Also, since the platform does not interface with or affect other plant systems or equipment, no new accidents or malfunctions are created. The consequences of previously evaluated accidents and malfunctions are not increased. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-046)

28. MR 98-116, (Unit 2), 4160 V.

The modification replaces Westinghouse 4160 Vac Model 50DH350 air magnetic breakers with Westinghouse 4160 Vac Model 50DH-VR350 vacuum breakers.

Summary of Safety Evaluation: The SE revision simply adds reference to ECR 99-0032 and Calculation 99-0011 that evaluated the seismic qualification of bus 2A-05. No other changes to prior evaluations exist. The modification does not pose a USQ nor does it require a change to the TS. (SE 98-111-01)

29. MR 98-117*A, (Unit 1), 4160 V.

The modification replaces Westinghouse 4160 Vac Model 50DH350 air magnetic breakers with Westinghouse 4160 Vac Model 50DH-VR350 vacuum breakers. The vacuum breakers are designed for use in original Model 50DH breaker cubicles. They provide equal or superior electrical and mechanical performance as compared to the original air magnetic breakers. The modification replaces breakers in buses 1A-01, 1A-02, 1A-03, 1A-04, and 1A-05, unless it requires a unit outage for replacement.

Summary of Safety Evaluation: The new vacuum breakers are an improvement in design and technology over the original air magnetic breakers. Vacuum breaker technology has proven reliable in industrial and utility applications. The new breakers are less subject to wear and are more easily maintained than the old breakers. For these reasons, the failure rate of the new breakers is expected to be as low as or lower than that of the original breakers. Therefore, the probability of the loss of power to safety-related equipment supplied by the vacuum breakers is not increased. Testing and qualification of the breakers ensures that no common failure modes exist.

The vacuum breakers are manufactured to specifications in IEEE C37.59-1991. The breakers are seismically qualified by testing and analysis in accordance with IEEE-344-1987. The seismic spectrum to which the breakers were tested envelopes the design basis earthquake for PBNP. Westinghouse performed final design testing on the first vacuum retrofit.

The replacement breakers have epoxy bushings that are less susceptible to moisture absorbency and chemical degradation than the ceramic bushing insulation on the original DH breakers. This results in a lower probability of primary insulation failure with the new breakers. The primary failure mechanism is unchanged. The inability to interrupt a current arc during contact separation remains the same. The failure in vacuum applications is localized to the contacts within the vacuum interrupter enclosure compared to a more catastrophic failure for air circuit breakers. Therefore, a failure of a new breaker results in less severe consequences than the failure of an original breaker. Other breaker failure modes, including failure to open or close when required, results in identical consequences to failures of the original air magnetic breakers.

The safety-related functions of the new breakers are the same as the original. Control logic functions remain the same. The vacuum retrofit is mounted on a breaker truck designed the same as the original air magnetic to maintain proper interface with the original cell components. The breaker racking mechanism interlocks are maintained to ensure that a breaker is not repositioned in the cubicle while the main contacts are closed. Shutter operation is accomplished in the same manner as with the original breaker/cell interface. The major difference between the new and original breakers is that the replacement breaker contacts are completely enclosed in a sealed vacuum enclosure, while the original air magnetic breaker contacts operate in ambient air accompanied by a blast of air from a puffer assembly. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-076)

30. MR 99-005, (Unit 1), Instrument Air (IA).

The modification replaces the valve trim in the IA containment isolation valves and slows the open time of the valves. The valves had quick opening trim. The modification installs equal percentage valve trim and slows the open time of the valve. The changes decrease initial demand on the instrument air system header, allowing for a smooth draw from the IA system. The open logic is also changed to allow the valve to continue to open after the selector switch is returned to auto.

Summary of Safety Evaluation: No accident or event has loss of IA as an initiator to an accident either directly or indirectly. The new trim pack is sized for full flow with only one valve operational. The function of the IA isolation valve is not changed. The new trim pack is sized for full flow with only one valve operational. The valve isolation characteristics are not changed by use of the new trim. Containment isolation is important to safety. The IA isolation valves are required to shut on a containment isolation signal. The CLB also states that the containment isolation valves automatically shut on loss of IA pressure. The fail position of these valves is not changed. Changes to the IA tubing supplying the isolation valve operator to facilitate slowing of the stroke open time does not cause the valve to remain open. The accumulator added between the solenoid and the operator has no moving parts to restrict air flow. The IA system is high quality, oil free air. Clogging of the needle valve or accumulator is unlikely. The valves shut time is increased slightly. The isolation valve shut time is not addressed in the CLB; however, the administrative limit of 1 minute is not violated (see NUREG SRP 0800). The new configuration allows the valve to stroke open after the selector switch in the control room is returned to auto. The new logic configuration does not cause the valve to change position without operator action when containment integrity is reset. This remains unchanged. The valve position indication light logic to the control room is not changed. The slowing of the open time improves the response of the plant.

The IA system is a common system, with interruption affecting the operating unit during a SI termination. The modification reduces the probability of a low IA header pressure at the MSIV. The modification does not change the capability of providing isolation during containment isolation and loss of IA pressure. The modification slows the shut time of the "fail close" containment isolation valve. However, the administrative limit of less than 1 minute is maintained. The modification does not affect the down stream containment isolation check valve from performing its safety-related function. Also, under normal conditions, the IA header pressure is always greater than the maximum accident containment pressure. Therefore, flow is into containment. The accumulator is adequately restrained to prevent it from affecting nearby seismic Class 1 or 2 equipment. Therefore, the modification does not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB. The modification does not change the availability of IA for normal operation or during outages. Changing the rate that the IA isolation valve strokes open causing IA to repressurize containment slower does not affect the basis for margins of safety. Reestablishing IA in containment is not required to mitigate design basis accidents. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-075)

Summary of Safety Evaluation: The SE revision clarifies precautions of installation testing restrictions with respect to the refueling unit and the operating unit. Installation occurs during a refueling outage. PMT is performed to verify containment integrity, stroke times and functional testing of the new configuration for at least one valve on each unit. The PMT includes verification of IA system response to expected conditions after containment isolation and bleed off of IA pressure in containment. Adequate precautions exist for both the unit in a refueling outage and operating unit or adequate precautions for a dual unit outage during PMT. The work plan verifies the manipulator crane and RCCA change fixture are not in use and LTOP is not required to be in service. The PMT is aborted if IA header pressure is below the expected design limits. Calculation 99-0081 verifies that the trim change functions as designed, and verifies the IA compressors are adequate for expected conditions. (SE 99-075-01)

31. MR 99-009*A, (Unit 1 and 2), Main Steam.

The modification removes the WABCO quick exhaust device from the Unit 1 and 2 condenser steam dump valves pneumatic control schemes. The WABCO devices are replaced with a volume booster that can be adjusted to port air from each steam dump valve in a more controlled manner without the system oscillations and valve cycling experienced with the WABCO device installed. The modification is installed with the affected steam dump valve removed from service. The modification does not require all eight valves to be modified prior to system restoration and reactor power ascension. However, the valves are modified as necessary during the startup to ensure that the reactor power level does not exceed the associated condenser steam dump capability.

Summary of Safety Evaluation: The volume booster does not affect the speed the valves open as required in the CLB, but it may cause the steam dump valves to modulate shut more slowly. The valves will, however, continue to operate within the bounds of design basis. No new failure mode is introduced by the modification.

Condenser steam dump failure was previously evaluated as an initiator for the excessive load increase accident. FSAR Chapter 14 events assume the condenser steam dump system do not operate during or after the transient since their operation minimizes the pressure transient or would not result in the most severe accident consequences. The condenser steam dump valves are not equipment important to safety. The modification does not affect the system as described in the basis of TS 15.3.4. The modification does not pose a USQ nor does it require a change to the TS. (SEs 99-019, 99-029)

32. MR 99-017, (Unit 1), Service Water.

The modification replaces a section of 6' SW return piping from the 1HX-23A&B turbine lube oil coolers. The original carbon steel piping is replaced with Type 304 stainless steel. The section replaced is downstream of control valve 1SW-2854 and extends from the downstream valve flange through the downstream elbow. The parts replaced include a 6" 90° elbow, a 16' 6" straight pipe section, a 6" to 4" reducer, 4" weld neck flange for the control valve including the studs and bolts, flexitallic gasket, a 1 x 6 sockolet and a section of the 1" drain pipe up to drain valve 1SW-454.

Summary of Safety Evaluation: The piping is off of a nonessential SW load. Installation occurs during a refueling outage with Unit 1 in cold or refueling shutdown with the SW supply to the Unit 1 turbine building isolated. The valves and line stops used to isolate the SW system have appropriate pressure and temperature ratings to adequately isolate the system and prevent leakage. Should leakage occur from upstream, flooding could occur in the turbine building. However, flooding in the turbine building non-seismic piping was analyzed and sufficient flood mitigation features exist to protect safety-related equipment. The leakage would not be greater than the flow assumed in the SW model for an operating unit, so flow to the essential SW loads would be maintained.

Failure of the isolation valves or line stops downstream have the potential to introduce air into the Unit 2 condensers. This could cause a loss of condenser vacuum and a unit shutdown, but would not present a significant degradation of the SW system. Approved procedures for loss of condenser vacuum define the required operational actions for this condition. The stainless steel replacement components are stronger than the original carbon steel components at the design conditions. The function and configuration of this portion of the SW system does not change, and the piping section is a return line from a non-safety related load. The portion of the SW system cannot affect the consequences of accidents evaluated in the CLB, and SW flow is maintained to essential loads. Flow to the lube oil coolers increase slightly with the decreased resistance of new piping, but does not exceed the 40 gpm of available leakage margin in the SW flow model. This is a reasonable determination based on review of the flow model and the small section of piping replaced. Testing quantifies affects on the SW flow model from the new piping. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-084)

Summary of Safety Evaluation: The SE revision further discusses flows with respect to the modification. Flow to the lube oil coolers increases slightly with the decreased resistance of new piping, but does not exceed the CLB analysis. This is a reasonable determination based on the small section of piping replaced and the results from similar work on Unit 2; therefore, the modification does not increase the radiological consequences of an accident, event, or malfunction of equipment as evaluated in the CLB. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-084-01)

33. MR 99-024, (Unit 1), Residual Heat Removal (RHR).

    The modification involves the permanent removal of two safety-related snubbers (AC-601R-3-R-350 and AC-601R-3-R-356) from the auxiliary coolant system/RHR piping. The removal of these snubbers can be performed at anytime and does not necessarily require a refueling outage. Piping Stress Report Accession WE-100112 identifies that both snubbers may be removed from service and still keep the piping within CLB acceptance limits.

    Summary of Safety Evaluation: The pressure boundary of the RHR system is not affected by the modification; therefore, the system remains functional throughout the work. There are no interim configurations to consider since stress report WE-100112 has qualified the piping system with or without these snubbers. Although no functional changes to the RHR system result from the modification, the removal of the snubbers from the plant requires a revision to FSAR Table 6.2-12, "Safety Related Snubbers."

    Since there are no functional changes to the facility, neither the probability nor the consequences of accidents or equipment malfunctions are affected. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-061)

34. MR 99-027, (Common), CVCS.

    The modification installs a tie-in between the T-8A-C CVCS holdup tanks and T-10A&B monitor tanks. It is periodically required to discharge the contents of the waste holdup tanks from a tank designated as a waste discharge tank. The monitor tanks are designated as such tanks. The tie-in is from the gas stripper feed line to the monitor tank line. The lines are both 1" stainless steel lines located in the PAB El. 26'. The modification requires approximately 6' of piping, a valve on the tie-in, a manual isolation valve, and various fittings. Liquid effluent in the holdup tanks is routed to the monitor tanks via the boric acid evaporator feed demineralizers for subsequent discharge. A holdup tank can be aligned in a recirculation lineup through the boric acid evaporator feed demineralizers and back to the holdup tank to allow for clean up and sampling. Once the contents of the holdup tank have been cleaned up sufficiently by the boric acid evaporator feed demineralizers, they can be transferred to one of the monitor tanks via this tie-in. After holdup water is transferred to a monitor tank, it can be returned to a holdup tank if necessary. The tie-in allows use of an existing flow control valve, 2FCV-159.

Summary of Safety Evaluation: The rupture of a holdup tank (HUT) and a CVCS malfunction are evaluated in FSAR Chapter 14. In the event of a CVCS malfunction, the part of the system modified is not necessary for the safe shutdown of the plant and could not cause an unplanned dilution of the RCS. With regard to HUT rupture, no volume is added to the HUTs as a result of the modification. The piping shall be designed and installed to meet or exceed original piping code B31.1. The tie-in will, therefore, not increase the probability of a HUT rupture. The pipe routing was evaluated and determined to meet or exceed the required seismic qualification. The tie-in connects two lines of piping that are non-safety related, not vital to the safe shutdown of the plant. Therefore, the operation of the valves to be installed is not important to safety. The portion of the system modified is completely isolable during the modification. Appropriate installation precautions are taken and post-modification testing is performed to ensure that new piping tie-in functions as intended. The modification does not increase the activity level of the holdup tank contents. The modification does not increase the amount of contaminated fluid in the HUTs, nor does it increase the activity level of the fluid passing through the tie-in. The piping system is not called upon to mitigate accidents, events, or equipment malfunction important to safety in the CLB. No new failure modes are added as a result of the modification. Therefore, the modification does not increase the probability of occurrence, possibility, or radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the CLB or otherwise. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-039)

35. MRs 99-060*A , 99-060*B, 99-061*A, 99-060*B, (Unit 1 and 2), CVCS.

The modifications make wiring changes to the control circuit of the CV-112C valve for each unit. The physical routing of the conductors are rerouted to eliminate the possibility of a spurious operation of the valves as a result of a postulated Appendix R fire.

Summary of Safety Evaluation: The function of the circuit is not changed. The change eliminates a possible malfunction method for the valves resulting in an improved circuit design that supports the 10 CFR 50 Appendix R requirements. Appropriate precautions exist to ensure that operating equipment can not be inadvertently operated while the work is in progress. Operations is alerted to the need to manually position the valve should it be required while work is in progress. Installation of the wiring and dedicated conduits are in accordance with applicable safety and seismic requirements for the PAB. Design and installation controls with testing following installation ensures valve operability prior to return to service. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-122)

TEMPORARY MODIFICATIONS

The following temporary modifications (TMs) were implemented in 1999:

1. TM 98-055, (Unit 2), Temporary Power to MCC 2B-31 During 2B-03 Bus Outage.

   The TM supplies temporary power to motor control center (MCC) 2B-31 while its normal supply, 2B-03, is inoperable during Unit 2 Refueling 23. The temporary power to 2B-31 occurs by installation of temporary cable that extends the power feed for power panel PP-18 to 2B-31. PP-18 supplies Unit 2 pressurizer backup heater Group D. The pressurizer backup heater Group D is not required to be in service during refueling shutdown conditions; therefore, it can be inoperable during this TM installation. PP-18 is powered from 2B-04, breaker 2B52-29B. The breaker has a setting of 350 A. The minimum size of the temporary cables is 250 MCM, which has an ampacity of 455A in free air (90°C insulation) per the NEC. MCC 2B-31 is rated at 800 A. However, several loads on 2B-31 are removed from service to ensure the 350 A temporary supply is adequate.

   Summary of Safety Evaluation: The TM affects 480 V power supplies PP-18, 2B-31, 2B-03, and 2B-04. Of these, PP-18 and 2B-31 are non-QA, non-safety related. Bus 2B-03 is not affected by the removal of 2B-31. While 2B-31 is temporarily supplied by 2B-04, isolation in the event of a short circuit is provided by breaker 2B52-29B in 2B-04. The temporary power supply to 2B-31 takes the place of PP-18, which is a normal load on 2B-04. The logic for stripping non-safety related load, including 2B-31, from 2B-04 in the event of an accident on either unit is unchanged. Therefore, 2B-04 has adequate capacity to supply MCC 2B-31 with adequate margin to supply safety-related loads required to be operable during the plant conditions in which the TM is installed. The change does not affect the functions of safety-related equipment or systems and does not remove safety-related equipment or systems from service. The temporary cables are not run through fire barriers and do not present intervening combustibles between safety trains. Installation of the temporary cables does not affect seismic supports nor exceed existing cable tray fill limits. There are no unusual system or equipment lineups or changes that could introduce the possibility of an accident or event of a different type than previously evaluated in the CLB. Possible malfunctions associated with this change include short circuits, open circuits, and improper breaker operation. These types of malfunctions are possible in the existing configuration. No new types of equipment or materials are installed by the TM. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-001)

2. TM 99-005, (Unit 2), Temporary Power to 2B-40 Supplied from 1B-40.

   The TM supplies temporary power to panel 2B-40 that is inoperable during the non-safety related portion of the 1B-40 work. The temporary power ensures heating and lighting is provided to the EDG building during cold weather conditions during 2A-06 work activities.

   Summary of Safety Evaluation: A temporary cable is installed from the non-safety related spare 100 amp breaker 1B52-409MR located in 1B-40 to the 2B-40 incoming line breaker 2B00-401M. Loading on 1B-40 is checked to ensure the total load does not exceed the rating of 1B-40. The current flow per phase is limited to 100 amps.

   The TM is for a short duration, less than 7 days. The temporary supply breaker on 1B-40 is stripped on a loss of voltage signal. Train separation issues do not exist with respect to 1B-40 and 2B-40 both being Train B safeguards. The cable is routed entirely within the EDG building. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-005)

3. TM 99-008, (Unit 2), Turbine Post Retrofit Test Instrumentation Installation and Removal.

   TM 99-008 addresses activities associated with the installation and removal of the temporary test instrumentation required to support performance of PBTP 095, "Turbine Post Retrofit Performance Test Unit 2." PBTP 095 collects secondary plant performance data following completion of the Unit 2 low pressure (LP) turbine retrofit (MR 95-004.) TM 99-008 is identical to TM 98-006 performed for Unit 1, except

TM 99-008 incorporates minor lessons learned during the Unit 1 work. Data collected from PBTP 095 is compared to that collected during the turbine performance test (PBTP 085) that was performed prior to the Unit 2 LP turbine modification. This assists with the determination of the overall performance gain.

Summary of Safety Evaluation: Temporary flow and pressure instrumentation is installed using tubing/pipe with pressure ratings equal to or greater than original plant design and fittings are pressure tested prior to placing it in parallel service with existing plant instruments to ensure an adequate pressure boundary. The flow and pressure instruments are connected to existing drains/isolation valves. Temporary temperature instrumentation is installed in existing thermowells. The temporary instrumentation is provided with independent power supplies and installed such that a failure of a temporary instrument does not cause or prevent the actuation of a control or protection signal.

Temporary main generator metering instruments are connected to existing plant instrumentation at interface points and the associated equipment is specifically designed to be used without interruption of the CT or PT signals that are commonly used during relay replacement and calibration. The generator output metering and protective relaying circuits are not safety-related and are utilized in the EH system as input to the load drop anticipatory circuit (closure of the governor valves via the overspeed protection circuitry if ~30% load mismatch is sensed between MW output and crossover pressure and the main generator breakers are open). The high accuracy transducer is a passive element in the circuit and does not produce feedback signals to other portions of the metering or protective circuitry such that a protective function could be disabled or a malfunction of equipment important to safety could occur. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-021)

4. TM 99-013, (Common), Troubleshooting the HOH Demineralizer for High Inlet Pressure.

Operations work plan (WP) 99-041 troubleshoots the high pressure source and installs a fabricated pressure gauge flange per TM 99-013. The TM isolates and partially drains the HOH mixed bed demineralizer and replaces the resin fill flange with one that has an attached pressure gauge. Prior to establishing flow to the HOH mixed bed, Operations diverts letdown flow to the chemical and volume control system (CVCS) holdup tanks and manually blends to the volume control tank (VCT) to maintain the desired VCT level. Flow through the HOH mixed bed is established by throttling the LiOH demineralizers outlet diaphragm valve. The demineralizer is then placed on line in parallel with the Train B demineralizer, while letdown is diverted to a CVCS holdup tank. Comparative pressure readings are obtained from the letdown line pressure indicator, deborating demineralizer pressure indicator, reactor coolant inlet pressure indicator, and the pressure gauge on the installed resin fill flange. The collected pressure data is used to determine the source of the unusual high inlet pressure to 2U-1A. Pressure on the bed is maintained around 100 psig to ensure that upstream diaphragm valves are not pressured to their 150 psi limit. The flanged pressure gauge is removed and the resin fill flange reinstalled after troubleshooting is completed.

Summary of Safety Evaluation: The temporarily installed pressure gauge flange assembly is built per PBNP design guidelines for CH-151R pipe class. The assembly is installed and visually inspected for leakage as required by construction Code B31.1 for non Section XI code class. Both the pressure gauge, vent valve, and associated tubing are rated for normal system temperature and pressure. The temporary pressure gauge flange is more than adequate for the pressure retaining boundary. The LiOH mixed bed outlet diaphragm valve is throttled to establish flow to the HOH mixed bed demineralizer. Pressure is maintained below the rated upstream diaphragm valves design pressure of 150 psi. The use of the valve as a throttle valve does not affect the function or operation of the valve since the design characteristics for diaphragm valves allows them to be positioned in an open, shut, or throttled position. Operations diverts letdown flow to the CVCS holdup tanks while blending to the VCT prior to placing the HOH demineralizer in service. Letdown is diverted in order for Operations to properly control the boron concentration in the RCS. The boron concentration in the HOH demineralizer had been higher than that of the RCS. Operations blends to the VCT as needed, to maintain VCT level and concentration at a desired level. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-022)

5. TM 99-015, (Unit 1), Install Blank Flange on W-003A Suction Duct, Secure CRDM Ventilation with RCS Temperature <350°F.

Unit 1 control rod drive mechanism (CRDM) shroud fan motor W-003A-M is replaced via WO 9817044. A section of the CRDM ventilation duct work on the suction side of the fan is disconnected and removed for disassembly of the motor from the fan. The work is performed with Unit 1 shutdown, but not in cold shutdown conditions. Shroud ventilation is maintained until the plant is in cold shutdown. CRDM shroud ventilation provides cooling to the CRDM coils that are located inside the CRDM shroud just above the reactor head. The ventilation is drawn into the shroud from above, through the CRDM penetrations, and passes down through the shroud. This removes heat from the CRDM coils. It then goes out and up through duct work to the CRDM shroud fans. The rod position indication (RPI) coils are outside the CRDM shroud, above the CRDM coils.

Summary of Safety Evaluation: Normal CRDM shroud fan configuration with the plant above cold shutdown is to have one shroud fan running with the other one secured. Therefore, it is desired to maintain the W-003B CRDM shroud fan running as much as possible during the work. The two CRDM shroud fans are supplied from a common duct ring header. With one fan normally running, reverse air flow through the idle fan is prevented by a backdraft damper mounted on the discharge side of the fan. Opening the duct at the suction side of W-003A has the potential to adversely affect W-003B fan operation for the following reasons: 1) Per the fan manufacturer there is potential to damage a fan motor if it is run with no inlet resistance. This could be the case if the W-003B fan is run with the W-003A fan suction duct open (a 28" diameter opening); 2) Leaving the duct open at the W-003A fan suction would likely cause a short-cycling of the CRDM ventilation, with air drawn in through the new duct opening, and blown out through the W-003B fan, instead of being drawn up from the shroud area, whose flow path has a greater system resistance; 3) There is a potential for FME intrusion while running the W-003B fan with the W-003A fan suction duct open and work being done in that area. For these reasons, a blank flange is installed to allow the W-003B fan to run during the repair of W-003A. The CRDM ducting, fan and motor are not safety-related. They are QA-related because they were constructed as seismic Class I. The blank flange is installed and supported to maintain its seismic Class I integrity.

The suction ducting for W-003A is in communication with the suction ducting for W-003B. This ensures that W-003B is secured during installation and removal of the blank flange for protection from overcurrent issues because of a dropped system resistance and potential introduction of foreign material. After repairs to W-003A are complete, the temporary blank flange is removed and the ducting restored to its original condition. Because CRDM shroud ventilation is secured during installation and removal of the blank flange, Unit 1 RCS is cooled down to <350°F to afford protection to the temperature sensitive CRDM and RPI components, taking no credit for CRDM shroud ventilation availability.

The CRDM ventilation system, the CRDM coils and RPI are not credited in analyzed events or accidents for mitigation or monitoring. The rod control cluster assemblies (RCCAs) are relied upon, but not the CRDM coils. TS definition of operability of the RCCAs is that the RCCA drops upon removal of stationary gripper coil voltage. Therefore, the TM does not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety previously evaluated in the licensing basis. The TM is performed so that temperature sensitive components are not damaged. Therefore, the probability of occurrence of a RCCA drop or other accidents or events previously evaluated in the CLB are not increased, and no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB will not occur. No new failure modes in plant systems important to safety are created. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-036)

6. TMs 99-024 through 99-029, (Common), Temporary Power to PP-54, PP-55, PP-63, PP-70, and B-81 Supplied from PP-4, PP-9, 2B-01, and 2B-41.

The TMs supply power to loads requested by Operations during the B-08/B-09 bus outage. The TMs supply temporary power to selected loads connected to B-81, PP-54, PP-55, PP-63, and PP-70 from 2B-01, 2B-41, PP-4, and PP-9 for approximately one week. Temporary breakers are installed to protect temporary cable where existing or spare breakers are not available. Temporary cables are sized per National Electric Code to coordinate with breaker size. B-81, PP-54, PP-55, PP-63, and PP-70 loads not requested by Operations are tagged out. Loads on PP-4, PP-9, 2B-01, and 2B-41 are tagged out to provide temporary power to B-81, PP-54, PP-55, PP-63, and PP-70 without overloading the supply buses. TMs are removed prior to removing tags on supply buses.

Summary of Safety Evaluation: Installation of temporary cables do not affect train separation. Cables are not run in conduit or trays. Penetration of fire barriers is in accordance with existing plant procedures so as to not increase the probability of an accident or fire.

B-81, PP-54, PP-55, PP-63, PP-70, 2B-01, 2B-41, PP-4, and PP-9 are not safety-related. The normal supply for B-81, PP-54, PP-55, PP-63, and PP-70 is shown on FSAR Figures 8.1 and 8.10. The loads on 2B-01 are shown on FSAR Figure 8.9. PP-54 is mentioned in FSAR Section 8.6.2, Figure 8.6-1, Figure 8.6-2, and Figure 8-10. By supplying temporary power to B-81, PP-54, PP-55, PP-63, and PP-70 from 2B-01, 2B-41, PP-4 and PP-9, PBNP is deviating from the facility as described on these figures. This change is for a short duration of approximately one week thus not necessitating a change to the FSAR.

FSAR Section 8.6.2 states that the backup power source for each instrument channel is from non-safety related Y-15 or Y-16 buses which are supplied from bus B-09 through PP-54.

SE 99-066 previously evaluated de-energizing Y-15 and Y-16. It concluded that the inverters or buses would not be rendered inoperable. Therefore, the TM does not pose a USQ nor does it require a change in the TS. (SE 99-054)

7. TM 99-032, (Common), Treatment of Service Water System with Chlorine Dioxide to Eradicate Zebra Mussels.

   Chlorine dioxide is produced by temporary equipment installed via TM 99-032 and injected into the service water (SW) pump bays via the normal chlorination system injection point. The injection treats the SW system for elimination of macrofouling in general and adult zebra mussels in particular. The treatment injects approximately 4 ppm of chlorine dioxide into the SW system for approximately three days. The treatment occurred in the third quarter of 1999.

   Summary of Safety Evaluation: Vendor supplied chemicals and equipment are placed south of the circulating water pumphouse, with PVC lines connected to the normal chlorination system inputs into the SW pump bays. A secondary chemical containment is placed to prevent potential spills from interacting with equipment or draining to Lake Michigan. The normal SW heat exchanger lineup is in use, with no planned alterations of the lineup during the treatment, except that the G-01 and G-02 emergency diesel generator (EDG) coolers and the auxiliary feedwater pump bearings have flow supplied to allow treatment of these items. The lineup is accounted for in the SW flow model and does not affect the operability of the equipment. Equipment parameters that may indicate blockage of components by zebra mussel shells are monitored, and contingency plans for blockage are written into a separate work plan. Monitoring of operating parameters during and after the treatment and inspections performed after the treatment assess the continued operability of SW equipment.

   The treatment can not initiate an accident or event, whether new or previously evaluate. Although there is a possibility to impact the operation of some safety-related equipment that use SW for cooling because of blockage by zebra mussel shells, the lack of zebra mussels found in the SW system despite periodic inspections makes this event improbable. In the unlikely event of a flow restriction in a SW component important to safety occurs, precautions are in place to diagnose and correct the condition prior to effect upon the operation or performance of either Unit. If required, the applicable TS required action is entered. Since this is the fourth such treatment, PBNP is well aware of what to expect and what could potentially occur. The addition of chlorine dioxide presents no radiological issues, and does not impact the radiological consequences of an accident. The temporary placement of chemicals near the circulating water pumphouse was evaluated and found acceptable with respect to control room habitability and our CLB. A sufficient margin of safety is maintained for SW flows by controlling the equipment that is in the service at any given time. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-092)

Summary of Safety Evaluation: The SE revision clarifies that the EDG fire water sprinkler lines are not part of the treatment nor the TM. This line was not treated during the 1997 evolution and inspections performed since then have provided no indications of shells in this area. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-092-01)

Summary of Safety Evaluation: The SE revision clarifies that EDG coolers are radiographed or cleaned for a three week period once the chemical treatment is completed. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-092-02)

8. TMs 99-032, 99-033, (Common), Treatment of Containment Fan Coolers (CFCs) with Rydlyme.

In the third quarter of 1999, PBNP treated the SW system with chlorine dioxide for the purpose of killing zebra mussels. A contingency plan was developed because of a concern that zebra mussel shells may detach from their present locations and possibly cause blockage in the CFCs. Since the major component of zebra mussel shells is calcium carbonate, an industrial descaler is used to dissolve shells that may block an affected cooler to the point of inoperability. The contingency plans for up to four of the eight CFCs to be fouled simultaneously. The equipment maintenance history of the CFCs shows that only the "C" and "D" CFCs may foul because of their position in containment. The "A" and "B" CFCs are over 60' above their supply headers, and the relatively heavy shells have not been known to be pushed up the piping to foul its tubes. The blockage is expected to be either biomass, which is dissolved by the chlorine dioxide, or is calcium carbonate based because of zebra mussel shells. If three or four CFCs in a unit are not operable at any given time, then the affected unit must shut down per TS 15.3.0.

Summary of Safety Evaluation: The affected CFCs are isolated, and a 330 gallon tank and a cart with a circulation pump are brought in with a chemical to clear the blockage. The pump draws suction from the tank and injects into the 1" local drain valve in the SW supply. The treatment occurs at ambient temperature, and the vendor pump injects 75 gpm at 60' of head, which is less than the 100 psig design pressure of the CFCs. The 1" drain in the return line deposits into the tank to allow the liquid to evolve carbon dioxide and be strained for solids prior to recirculation.

Since the shells are made largely of calcium carbonate, the same material as the scale that forms on pipe walls, and industrial descaler is used to dissolve the shells. The chosen descaler is Rydlyme, which destroys shells within six hours at a 50% concentration and is benign to copper, carbon steel, stainless steel, and other materials that Rydlyme is expected to come in contact. It is biodegradable and has no toxic fumes. Carbon dioxide gas is evolved with Rydlyme reacts with scale. Since carbon dioxide is known to be non-toxic, non-flammable, and non-corrosive, the only concern is to ensure enough ventilation is available for the expected amount of evolved gas to dissipate. Since only a small amount of shells are expected, the amount of gas evolved is minimal and within the capacity of local ventilation. Since the CFCs are considered inoperable and isolated before Rydlyme is applied, no accident or event can be initiated. Since only the SW side of the CFCs is affected, there are no radiological concerns. The SW system is considered a closed system inside containment, and the valves are shown on FSAR Figure 5.2-35 as containment boundary valves. The outside containment 1" drain valves for the SW side of the CFCs are open and controlled by a Level 2 dedicated operator as allowed by TS 15.3.6.A.1.b.(1).(a).1, and controlled by OM 3.26. Even though the chemical is environmentally safe, the CFCs are flushed prior to return to service and the chemical waste liquid disposed to the retention pond. If the blockage is related to a source that is not affected by Rydlyme and not cleared within the allowed outage time, the affected unit performs the necessary shutdown. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-091)

Summary of Safety Evaluation: The SE revision was made to clarify vendor equipment location during this evolution. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-091-01)

Summary of Safety Evaluation: Following the treatment with Rydlyme, the CFC is flushed to the turbine hall sumps until the concentration of Rydlyme is reduced to less than detectable and therefore stop the increased corrosion rates due to the Rydlyme prior to returning the CFC to operation. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-091-02)

Summary of Safety Evaluation: Rydlyme is injected into the 1" inlet drain and lake water is discharged from the 1" outlet drain to the turbine hall sump pumps until an adequate concentration exists in the CFC. At that time, the Rydlyme solution is placed in circulation through the CFC. Rydlyme at a 50% concentration dissolves scale and shells completely in less than six hours. A spill of the waste chemical is not reportable unless the total amount discharged to the lake is greater than 5000 lbs. pure hydrochloric acid, or 50,000 lbs. of pure Rydlyme. In the improbable event that Rydlyme enters containment, it drains to the containment sump. The Corrosion Handbook by Uhlig indicates a general corrosion rate for stainless steel to be 0.065 ipy or 0.7 mils for 10% HCI for the 100 hour time restrictions suggested. Corrosion Resistance Table 4th Edition by Philip Schweitzer indicates the packing materials and non-metallic gasket materials are resistant to hydrochloric acid. The Metals Handbook 9th Edition indicates a corrosion rate for aerated 20% HCI of 50 mils/year. For a maximum exposure of 100 hours, this equates to 0.5 mils of material lost from the copper tubes of the CFCs that has a thickness of 45 mils. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-091-03)

9. TMs 99-037, 99-038, (Common), Containment Fan Cooler (CFC) Reverse Flushing.

A chlorine dioxide treatment to remove zebra mussels from the SW system resulted in the plugging of several of the heat exchanger tubes inside some of the CFCs. These TMs facilitate contingency plans to clean the CFC tubes for both units by forcing SW flow backwards through the CFCs. The work is done with the reactor shutdown. Other TMs may be installed in the future as needed to reverse flush the CFC tubes.

Summary of Safety Evaluation: The SW system is aligned such that two CFCs are inoperable for the unit, and two are inservice with SW flow and electrical power available. The common outlets to the CFCs are isolated which forces water backwards through one CFC and out a hose connection on the inlet check valve. Flushed debris is collected in the container where it can be visually inspected. The excess water is allowed to drain out the container and into the plant storm drains. This discharge is continuously monitored for radioactivity during flushing by connecting to RE-216, CFC liquid monitor. If RE-216 alarms, flushing is stopped. The RCS for the unit is <200°F during this evolution, and two CFCs (one from each train) are available during this operation. The outlet MOVs (SW-2907/2908) and the normally open outlet valve (1SW-144 or 2SW-273) are shut to provide maximum flow to the CFC during reverse flushing. The MOVs may be opened as necessary from the control room to provide increased cooling flow to the two inservice CFCs. Also, additional SW pumps may be started to increase the SW header pressure to provide for more effective flushing.

Loss of containment cooling cannot cause an accident or event. The other unit while operating will require service water, but loss of service water will not cause an accident or event. Should leakage occur from a hose fitting or a rupture of the hose occur, flooding might occur in the PAB or one of the facades. Flooding in the PAB or facades from non-seismic piping has been analyzed, and sufficient flood mitigation features exist to protect safety-related equipment. Loss of an Appendix R fire barrier does not constitute an accident or event. The activity does not increase the likelihood of the occurrence of a fire. The potential for an unmonitored radiological release is reduced by connecting the reverse flush discharge to the CFC liquid monitor (RE-216) to provide a continuously monitored flow path. This reverse flushing operation is done in accordance with the design ratings of the SW system, and the design pressure of the system is not exceeded. None of the safety functions of the inlet check valve are needed with the unit in a condition with the RCS temperature <200°F. This check valve is changed as part of TM installation, but is restored after the flushing is complete. The inlet to the liquid monitor sample line is facing away from the flushing flow to reduce the likelihood of blocking the supply lines to the monitor with debris from the CFCs. Also, this monitor path is valved in just after flushing flow has started to reduce the amount of air intake into the monitor. Safety-related electrical equipment located beneath the hose route does not fall if hose leakage occurs. The activity does not reduce the SW system's ability to remove decay heat from either unit, and the reverse flushing flow rate is within the SW model limits so the operating unit does not enter an LCO. This flushing improves the CFCs ability to remove heat from containment during accident conditions. The CFC is inoperable during the flushing, but two other CFCs (one from each train) are available. The flush water is discharged down the plant storm drains, and this flow is monitored using RE-216. The flushing is stopped if RE-216 alarms. Therefore, the activity does not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety as evaluated in the CLB. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-109)

10. TMs 99-040, 99-041, (Common), Differential Pressure Indicators for SW-02911 and SW-02912-BS.

The TM adds differential pressure indicators (DPIs) to provide accurate differential pressure readings across the North and South SW header Zurn strainers, SW-02911/02912-BS. The change ensures that the SW system can supply required flows, as analyzed in the service model, to equipment important to safety. The DPIs are added to the existing test connections on the lines for the North and South Zurn strainer DPI switches, DPIS-02911/02912, as shown on FSAR Figure 9.6-1. DPIS-02911/02912 and PI-02978/79/80/81 are used to initiate backflushing of basket strainers BS-029011/12. They are isolated during installation of the new DPIs. Basket strainers BS-02911/12 are continually backflushing during the installation of the new DPIs to ensure proper straining of the SW.

Summary of Safety Evaluation: The original backwashing controls initiated backwashing automatically when the differential pressure reaches 5 psi, or four hours after the preceding backwash cycle, whichever occurs first. The change ensures that the SW system can supply required flows to equipment important to safety. The DPIs are added to existing test connections on the lines for the North and South Zurn strainer DPI switches, DPI-02911/02912, as shown on FSAR Figure 9.6-1.

The DPIs are connected with flexible stainless steel tubing and compression fittings. The DPIs, as well as the other wetted equipment used in the installation, have an operating pressure equal to or greater than the SW system operating pressure of 100 psig. The equipment is secured in a temporary manner, such that its movement is restricted during a seismic event, and it is incapable of impacting surrounding equipment. The flexible stainless steel tubing absorbs vibration, or slight movement, which may occur during a seismic event. The use of compression fittings minimizes the possibility of a SW leak.

The DPIs use existing pressure taps to provide a local strainer differential pressure indication. The activity ensures that the entire SW system operability can be verified, and thereby ensures that the SW available to those components in an accident would not be changed based on CLB and TS requirements. The DPIs and associated tubing meets or exceeds the pressure requirements for the system. It performs no active function other than differential pressure indication. It is not credible that a loss of SW will occur during this installation. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-108)

11. TM 99-048, (Unit 1), Emergency Water Supply for Vacuum Priming Pump P-041.

The SW tagout for Unit 1 Refueling 25 isolates SW from the common P-041, the common vacuum priming pump. In order to provide redundancy for the operating unit, a source of water (approximately 12 gpm) needs to be temporarily attached to the inlet piping so that the pump can perform its backup duties. A hose is routed from a local demineralized (DI) water supply for this purpose. It is connected to DI-00091 only when the water supply is required. A check valve is installed in the hose near the DI valve to prevent back flow and an isolation valve is installed at the end of the hose just before the attachment to the clean out of Y-strainer YS-02983, normal water supply for the pump. The vacuum primping pump inlet SW valve, SW-00445A, is shut; therefore, the system is therefore isolable and not cross connected.

Summary of Safety Evaluation: TM 99-048 provides water to the common P-41 vacuum priming pump. This is necessary because of the unavailability of the normal water source (SW) for this pump during outage activities. The water supply consist of a hose attached to DI-00091 and runs overhead to the clean out of Y strainer (YS-02983) on the inlet side of P-41. Water is the medium used by this pump to effect the required function of the pump (vacuum to the CW system). A check valve at the water source and an isolation valve at the pump ensures no system cross connection concerns exist. Review of the overhead route for the hose indicates no challenges to the structural integrity of pipe hangers or other supports, nor does it create personal safety concerns.

The vacuum priming pump(s) are not initiators of design basis accidents or scenarios. They provide a means to maintain circulating water level in the condenser water boxes and the condensate coolers. As such, no CLB challenges are created by the operation of this pump. Providing for water at the rate of 12 gpm also does not challenge the DI water system. Provisions are addressed for concerns regarding the cross connections of systems via the use of check and isolation valves. The TM is for use on an "as needed" basis at the discretion of the DSS to augment the operating unit's vacuum priming pump. Work orders are in place for the installation and removal based on the SW system availability. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-118)

12. TM 99-054, (Common), New Fuel Elevator Hoist Limit Switch Adjustment and Mechanical Stop.

    TM 99-054 modifies the limit switch on Z-069, new fuel elevator (NFE) hoist. It had been set to allow the elevator to travel to the surface of the spent fuel pool (SFP). The elevator is temporarily used to replace top nozzles on irradiated fuel assemblies that are placed back into the Unit 1 core as part of Unit 1 Refueling 25. The limit switch must be reset to prevent lifting of the irradiated fuel assembly to the surface for radiological reasons. In addition, a temporary mechanical stop is put in place as a redundant measure in the event the limit switch fails to provide adequate time and acceptable radiological conditions to implement corrective actions. The SFP liner at the weld locations is examined to verify that it is returned to its original condition.

    The SE also addresses changes to RESP 2.3, "Defective Removable Top Nozzle Replacement." RESP 2.3 governs the use of Westinghouse vendor procedures to replace defective removable top nozzles, and includes the necessary precautions and compensatory measures. The replacement top nozzles are of the same design as those currently installed.

    Summary of Safety Evaluation: The performance of top nozzle replacement, as governed by RESP 2.3, is similar to other fuel manipulations performed at the plant. Like other fuel manipulations performed at the plant, manipulation of fuel in the SFP for nozzle replacement is procedurally controlled and the radiation levels continuously monitored. Compensatory measures for an unexpected loss of SFP level or an increase in radiation levels are provided in RESP 2.3. Engineering controls (such as handling tool length, limit switch and mechanical stop) are also applied during this activity. No fuel damage or criticality is expected to occur during the top nozzle replacement activities because of administrative and engineering controls applied during the activity. Adequate shielding depth is maintained.

    The change does not increase the probability of occurrence of an accident or event, or increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. The change does not create new failure modes or new accident initiators. There is no increase in the radiological consequences of an accident. The change does not increase the probability of occurrence of an accident or event of a different type, or increase the probability of occurrence of a malfunction of equipment important to safety of a different type than previously evaluated in the CLB. The TM does not pose a USQ nor does it require a change to the TS. (SE 99-125)

13. TM 99-056, (Unit 2), Furmanite Repair of 2MS-00244, the 2MS-02015 Atmospheric Steam Dump Inlet Isolation Valve.

    The change is an on-line leak sealant repair of 2MS-00244, the 2MS-02015 atmospheric steam dump inlet isolation valve. The leak is at the body-to-bonnet joint of the valve. The valve can not be isolated from the steam generator to perform repairs; therefore, the body-to-bonnet joint is sealed using an on-line leak sealant process. The sealant process utilizes a technology of injecting a sealing compound at high pressure to fill the leak volume. The technique was successfully performed on other components at PBNP, and the compound is acceptable for use on corrosion-resistant alloys per NP 3.1.1, "Chemical Contamination Control for Corrosion Resistant Alloys." 2MS-00244 is an ASME XI, Class 2 manual valve and is normally open. The atmospheric steam dump valves 2MS-02015 and 2MS-00244 are tested for return to service. TM 99-056 is removed when the unit is in its next cold shutdown condition, at which time valve 2MS-00244 will then be permanently repaired.

    Summary of Safety Evaluation: The injection of leak sealant compound has shown to not adversely affect 2MS-00244, and the downstream atmospheric steam dump valve 2MS-02015. Calculations show that at no time during the leak sealant procedure will the body-to-bonnet bolts be over stressed. The additional weight from the increased size of the injection nuts, injection washers, and the leak sealant compound is negligible when compared to the over all weight of the valve, and does not affect the seismic rating of the valve. The leak sealant is injected 180° from the location of the leak and then works it way towards the leak. The method ensures that the sealant fills cavities evenly prior to injection at source of leak. In addition, the body-to-bonnet gasket is a metal jacketed spiral round gasket that acts as a barrier to contain the leak sealant compound. Also, the amount of leak sealant compound physically injected into the leaking joint is limited and controlled by approved vendor procedure N-99281. These controls, as well as the existence of the metal jacketed spiral wound gasket that will act as a barrier to contain the leak sealant compound, minimizes the chance that the leak sealant compound migrates into the steam space of the valve. To further ensure that operability of the atmospheric steam dump valve 2MS-02015 is maintained, it is stroke tested after the temporary leak repair is complete.

The actual process of injecting the leak sealant compound has shown no increase in the probability of a steam pipe rupture or failure of the atmospheric steam dump valve to stroke properly. The atmospheric steam dump path is used in the steam generator tube rupture (FSAR Chapter 14.2.4) and rupture of a steam pipe (FSAR Chapter 14.2.5) accident analysis. Also, the operability of the atmospheric steam dump path is verified after the temporary repair is complete. Therefore, the radiological consequences of an accident are not increased. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-130)

Summary of Safety Evaluation: TM 99-056 is removed while the unit is in cold shutdown, at which time valve 2MS-00244 is permanently repaired. Calculations show that the stresses in the body-to-bonnet studs, as a result of the system pressure acting upon a larger surface area, exceed allowable stresses for the existing A193 B7 stud material. Therefore, new studs made of A540 B21 Class 1 material are used. After the on-line leak sealant repair is complete, the stresses developed in the new stud material are much less than the allowable stresses. Calculations show that at no time during the leak sealant procedure will the body-to-bonnet studs exceed their Code allowable stress. The additional weight added to the valve from the injection compound, the slightly longer new A540 B21 Class 1 studs, and injection hardware are negligible and do not adversely affect the seismic rating of the valve or the installation. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-130-01)

14. TM 99-059, (Unit 1), Stroke Testing Valve 1RH-700.

1RH-700 must be stroked to distribute packing adjustment forces throughout the packing material in order to effect an adequate seal such that leakage may be stopped. Due to RCS conditions, a low pressure permissive interlock must be jumpered in 1C-03 to allow stroking of 1RH-700.

Summary of Safety Evaluation: 1RH-700 must be stroked to distribute packing adjustment forces throughout the packing material in order to effect an adequate seal such that leakage may be stopped. RCS conditions require a low pressure permissive interlock to be jumpered in 1C-03 in order to successfully stroke 1RH-700. The low pressure permissive interlock prevents an operator from inadvertently overpressurizing RHR because of manual action. Administrative controls ensure that 1RH-701 is shut and remains shut while 1RH-700 is stroked. Per OP-7B, 1RH-701 is shut with power supply breaker isolated in an open position during plant startup. This ensures that the RHR system can not be subjected to an inadvertent overpressurization. Further, the RHR system has two relief valves installed to prevent the RHR system from overpressurizing. The relief valves are located just down stream of the 1RH-701 valve.

The 1RH-700 and 1RH-701 valves do not receive auto-close or auto-open signals. Heightened operator attention to the fact that 1RH-701 must be maintained shut is considered equivalent to the 1RH-700 low pressure permissive interlock. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-131)

15. TM 99-060, (Unit 1), 1SI-0853C Low Head Safety Injection (SI) Core Deluge Check Valve Sealant Repair.

The change is an application of on-line leak sealant to the body-to-bonnet joint of 1SI-00853C to isolate leakage. The valve can not be isolated from the RCS to perform repairs; therefore, the body-to-bonnet joint is sealed using an on-line leak sealant process. The studs and nuts are replaced one at a time with studs and nuts made of SB-637 UNS# N07718 material that is resistant to boric acid corrosion. The sealant process uses a technique of injecting a sealing compound at high pressure to fill the leak volume.

Summary of Safety Evaluation: 1SI-00853C is an ASME XI, Class 1 check valve. The TM is to be removed and the valve repaired during the next Unit 1 entry into cold shutdown with the RCS drained to 3/4 pipe or the next Unit 1 refueling outage. The review of the loss of coolant accident scenario and the compensatory measures planned during the TM are in compliance with PBNP procedures, design guidelines, and ASME Section XI requirements. The review of the seismic considerations, body-to-bonnet stud stresses, effects on the intended function of the valve, system effects, and TS impacts reveal that the TM does not adversely effect the component, system, or system function and availability. Thus, the TM does not challenge the safety-related functions of the RHR/SI systems that are to deliver borated cooling water to the RCS during the emergency core cooling system (ECCS) injection phase, to recirculate and cool water from the containment sumps and return it to the RCS during the ECCS recirculation phase, and to provide containment and RCS pressure boundaries. Therefore, the original assumptions for system performance during a LOCA scenario remain valid. The TM does not pose a USQ nor does it require a change in the TS. (SE 99-133)

Summary of Safety Evaluation: The SE revision addresses the stud/nut replacement, leak sealant process, including peening of the sealant clamp and the duration of the temporary installation (one refuel cycle maximum). The TM is to be removed and the valve repaired during the next Unit 1 entry into cold shutdown with the RCS drained to 3/4 pipe, or the next Unit 1 refueling outage. The TM and this SE evaluation do not pose a USQ nor does it require a change in the TS. (SE 99-133-01)

MISCELLANEOUS EVALUATIONS

The following evaluations were implemented in 1999:

1. CR 99-0994, Segregation of Site Security Force Function from Site Management Program Surveillance Functions for the ISFSI.

   The ISFSI SAR indicates that onsite Security is responsible for the maintenance surveillance as well as the facility security surveillance. However, we comply with the maintenance surveillance requirement by designating site personnel to perform the function and not having site Security perform the task. ISFSI SAR 11.1.2.2 and 11.2.9.2 indicate that Security is the detection mechanism for air vent blockage; however, Security functions only as the facility security surveillance force that ensures the integrity of the security fences and that safeguards is maintained.

   10 CFR 72.48 Evaluation Summary: Compliance with the requirements of the ISFSI SAR, SER and C of C are maintained because the maintenance surveillance and security surveillance requirements are performed by designated site personnel. Currently, the daily maintenance surveillance requirement is fulfilled by Radiation Protection personnel and the facility security surveillance is performed by Security personnel. The change is in title only, not functions performed. No USQ, significant increases in occupational exposure, significant unreviewed environmental impact, nor changes in the license conditions of the C of C are created. (SE 99-134)

2. CR 99-2180, Decrease Service Water Operating Temperature Upper Limit to 65°F.

   The SW intake temperature is changed from 80°F to 65°F in SW-related procedures and documents (e.g., OI-70, "Service Water Operation;" PBF-2033, "Safeguards Shift Logs;" and, PBF-2509, "Service Water Temperature Calculation"). The change is required because of the temperature restriction imposed by operability determination OD-99-2180. It is anticipated to last until May 2000. The document changes facilitate status control.

   Summary of Safety Evaluation: Limiting inoperability of the automatic isolation valves ensures that the design function of the system is maintained. Automatic isolation is a design feature of the SW system. Loss of SFP cooling is an anticipated plant response to an SI signal and is not an accident initiator. The SFP operates intermittently or continuously whenever there are spent fuel assemblies in the SFP dependent on heat removal requirements. In the event of complete failure of the cooling system for a long period of time, the SFP water inventory can be maintained with fire suppression system water. Limiting heat sink temperatures are within the existing normal operating band and do not increase the probability of a malfunction of equipment. The change does not pose a USQ nor does it require a change to the TS. (SE 99-129)

3. CRs 99-2180, 99-2235, Use of Local Isolation Valves as Containment Boundaries.

   The activity isolates a leaking cooling coil on the Unit 1 Train A and Train B and Unit 2 Train A containment fan coolers (CFCs) by shutting its associated inlet and outlet isolation valves, and opening the associated vent valve. The isolation of the coil results in 32 tubes being removed from service. The total number of tubes in each of the CFCs to be removed from service because of a combination of isolation and blockage (as indicated by thermography) are not permitted to exceed the number of tubes required to be in service to ensure operability of the CFC. The isolation valves are leak tested prior to taking credit for the system (and containment) as a pressure boundary.

   Summary of Safety Evaluation: Isolation of the coil does not have a negative affect on other systems or components supplied by SW. Containment integrity is maintained. Reduction in heat removal capability was evaluated via CR 99-2180 and is acceptable. The change does not pose a USQ nor does it require a change to the TS. (SEs 99-068, 99-105, 99-107)

4. DCNs 99-0399, 99-0402, 99-0403, 99-0406, 99-0407, and 99-0408, FSAR Figure Changes.

The changes clarify drawings associated with the reactor coolant, RHR and containment spray systems. The changes ensure that the affected drawings reflect the correct plant configuration of both units. Changes include showing SC-958 as normally shut; removing the RHR return line from "B" high head safety injection line; and, a capping a connection downstream of SI-863A.

Summary of Safety Evaluation: The drawing changes do not change physical valve positions, system functions or operation. Changes ensure the drawings reflect the correct configuration. Because there is no change in system function or operation, there is no change in system response to accidents or malfunctions stated in the FSAR. Also, since operation of the respective systems is unchanged, there is no possibility of an accident or malfunction of safety equipment of a different type than analyzed in the FSAR. The change does not pose a USQ nor does it require a change to the TS. (SE 99-018)

5. Extended Fuel Cycle Implementation, 18-Month Nominal Cycle Length.

The SE evaluates the implementation of extended fuel cycles (nominal 18 months) for Unit 1 and 2 starting with U2C24 and U1C26. A separate SE is performed for each cycle to evaluate the nuclear safety aspects of the cycle-specific core design and operation (e.g., rod worths, enrichments, IFBAs, loading pattern, shutdown margin, accident analyses, etc.) and compliance with TS for the core design aspects of each reload. Three potential impacts of the extension of the fuel cycle length from a nominal 12 month to a nominal 18 month cycle were identified. These are increased boron concentrations for reactivity control, increased surveillance, calibration, test, and maintenance intervals due to the longer period between refueling outages, and revised radiological source term and releases due to the higher uranium fuel loading, burnups, and capacity factors.

Summary of Safety Evaluation: Three potential impacts of extended fuel cycles were evaluated as follows: 1) Increased Boron Concentration: The increased TS boron concentration requirements for extended fuel cycles approved by the NRC via Amendments 180/190 were evaluated for the impact of chemistry changes on normal operation and accident mitigation, particularly the effectiveness of containment spray for scrubbing and retention of gaseous iodine following a design basis LOCA. The evaluations showed no adverse impact on the reactor fuel, reactor coolant system components including Alloy 600, nor on the effectiveness of containment spray; 2) Increased Surveillance, Calibration, Test, and Maintenance Intervals: These intervals were evaluated for extended fuel cycle operation. The results indicate no expected adverse impact on the reliability or performance of systems or components that could initiate or mitigate accidents, events, or malfunctions of equipment previously evaluated in the CLB. In addition, an evaluation of LERs at sister plants indicates no new type of accident, event, or malfunction of equipment could be created by the schedule extensions resulting from extended fuel cycles. TS requirements for calibration of instrumentation, electrical relays, and miscellaneous equipment at intervals no longer than 18 months must be met or the TS revised. NRC commitments regarding reactor trip breaker maintenance, EQ replacements, ECS system leak checks, specific valve cycling and tests, CCW heat exchanger cleaning and inspection, and AMSAC and emergency power tests must be performed at the committed intervals or the commitments appropriately revised; 3) Revised Radiological Source Term: The radiological source term in the reactor core for extended fuel cycles was compared to previous cycles and has not increased. In addition, the radiological releases offsite and control room does for design basis accidents were analyzed using source terms from a nominal 18 month cycle reload core design and the results reviewed and accepted by the NRC via Amendments 174/178.

Operation of extended fuel cycles does involve several TS changes related to specified maximum calibration intervals of 18 months for certain instruments and relays and for snubbers, but cycle operation is consistent with the TS provided the specified calibrations and surveillances are performed or the TS changed to extend the intervals. The change does not pose a USQ nor does it require a change to the TS. (SE 99-008)

6. FPEE-1999-016, Fire Separation of the Cable Spreading Room and Control Room Via a 2-Hour Fire Rated Concrete Block Wall.

According to the Fire Protection Review, dated June 1977 (1977 FHA), PBNP committed to upgrading the cable spreading room (CSR) boundary to achieve a 3 hour fire resistance. The commitment includes the hollow concrete block (HCB) walls surrounding Stair No. 33 at control building El. 44' that separates the control room (CR) from the CSR. MR 84-34 upgraded these walls to the committed 3 hour fire rating by applying a 3/4" thick gypsum plaster coat on the CSR side of the walls. However, a portion of the CSR north wall, referred as a Segment 115/23 that is located behind the F-11 plaster board enclosure, was omitted. The resulting fire resistance of this portion of the wall is 2 hours per U.L. Design No. U906.

FPEE-1996-016 evaluates the acceptability of this configuration. The conclusion is that the as-built configuration of this wall segment provides adequate separation of the CSR and CR and is therefore acceptable. A change to FPER Section 5.6 (Fire Zone 318) documents the acceptance of this 2 hour fire rated barrier.

Summary of Safety Evaluation: According to the results of FPEE-1999-016 fire barrier Segment 115/23 satisfies the design requirements of a 2 hour fire rated barrier per U.L. Design No. U906. The FPEE further concludes that based on the in-situ fire hazards in these rooms, the as-built configuration of this wall segment provides adequate separation of the CSR and CR and is therefore acceptable. The change to the FPER that accepts and documents the as-built configuration and fire rating of this wall does not challenge either the conclusions of the SSA or CR habitability. Therefore, the current accident analysis contained in the CLB remains current and unchanged. The change does not pose a USQ nor does it require a change to the TS. (SE 99-119)

7. FPER, Fire Protection Evaluation Report, August, 1999 Revision.

The FPER changes are documented on Fire Protection Evaluation Report Change Request forms. The FPER changes bring the program into current status. Also, the information or NRC exemptions obtained or granted are included in the Fire Protection Program. This is an enhancement to the fire protection program.

Summary of Safety Evaluation: The FPER changes resolve identified conditions requiring changes to make the document accurate. Evaluations ensure that the design basis for fire protection is not compromised. The reviews revealed that either minor adjustments within the program itself are necessary, including implementing procedures, or the changes are viewed as an enhancement to the program by introducing further clarification. Therefore, the equipment important to safety is not adversely affected by the FPER changes. The change does not pose a USQ nor does it require a change to the TS. (SE 99-097)

8. FPER Change Request 99-006, Fire Protection Engineering Evaluation (FPEE).

A condition was identified involving the presence of unprotected telephone wires and flexible light cords in the cable spreading room. This does not present a significant threat to safe shutdown components or the capability to safely shutdown the plant. The telephone wires and flexible fluorescent light fixture power cords that are necessary for the normal operations of the plant are addressed and are not combustible pathways across the 20' separation area. This is not intended to change or alter the exemption granted.

Summary of Safety Evaluation: The functions of both the telephone wires and the light cords are not independent not only from one another but also have no direct interface with other components within the room. A 20' separation area in the middle of the room is maintained by keeping the amount of combustibles isolated or minimized which is equivalency of a raceway enclosure in space, for the protection of exposure of intervening combustibles between Unit 1 and Unit 2 sides of the room. Should the light cords fail and ignite, their failure could result in the insulation burning in a vertical direction. However, the cords essentially pose an insignificant threat as an intervening combustible pathway because they do not leave the 20' separation area. The overall amount of exposed combustibles is greatly reduced and with the plant's defense-in-depth concept by using a detection and Halon suppression system within the room that will easily alert and extinguish a fire.

The routing of cables in either conduits or covered cable trays has been acceptable in the past. This isolates the exposure of intervening combustible pathways for the cable spreading room. NRC letter dated July 3, 1985, accepted the use of metal cable tray covers as a method of retarding fire propagation.

The plant has been evaluated and analyzed to confirm that the plant has the capability of an alternate safe shutdown method even with the complete loss of this room. AOP-10A provides procedural guidance to achieve and maintain safe shutdown in the event of a cable spreading room fire. The plant is capable of achieving and maintaining safe shutdown per the requirements of Appendix R. The change does not pose a USQ nor does it require a change to the TS. (SE 99-047)

Summary of Safety Evaluation: The initial deficient condition involving a bundle of telephone wires was brought into a designated configuration and the fixed combustibles for the cable spreading room have been re-evaluated and found acceptable therefore, the compensatory twice per shift fire rounds should be removed. The change does not pose a USQ nor does it require a change to the TS. (SE 99-047-01)

9. FPER Figures 5.5-2, 5.5-19 and 4.1.3-2, Revision to Wisconsin Electric drawing PBC-218.

The SE addresses FPER figure changes to reflect the purpose of plant fire rated barriers. Reclassification of the fire zones to non-fire rated barriers or Appendix A/industrial fire barriers, rather than Appendix R fire rated barriers does not require physical plant change.

Summary of Safety Evaluation: The barrier segments reclassified were identified as Appendix R fire rated barriers; however, these barriers are not required in support of the Appendix R safe shutdown analysis. Classification as Appendix R barriers requires these walls and their components to be more rigidly controlled than other plant fire barriers. While this level of control is justifiable for barriers that provide separation of safe shutdown equipment, it is not reasonable for fire barriers that provide property protection. The reclassification requires that these barriers be maintained in accordance with the appropriate NFPA and Wisconsin Administration Code requirements, but are not subject to fire protection TS limitations identified in FPER Section 7.2.

The safe shutdown analysis does not require these barriers for train separation of Appendix R safe shutdown equipment or cabling. In the event of an Appendix R fire, safe shutdown of the plant is achievable independent of the performance of these barriers. The reclassification of these fire barriers should be updated to reflect their current design basis as defined by the current safe shutdown analysis. The change does not pose a USQ nor does it require a change to the TS. (SE 99-062)

10. FPER Figure 6.6-6c, Revision to Westinghouse Drawing 110E029, Sheet 3.

The SE addresses a drawing change notice to make corrections to Westinghouse drawing 110E029, Sheet 3 to show component cooling system flow indicating switches, 2FIS-00649 and 2FIS-00650, as being connected electrically vice mechanically as they are currently represented. The drawing is also included in FPER Figure 6.6-6c.

Summary of Safety Evaluation: 2FIS-00649 and 2FIS-00650 are not safety-related nor are they important to safety. Accidents or events that might occur to the component cooling system have either been analyzed in the CLB or are bounded by accidents already evaluated in the CLB. Showing the flow indicating switches are electrically vice mechanically connected does not create the possibility of accidents or events of a different type as currently bounded by existing analysis. The change does not pose a USQ nor does it require a change to the TS. (SE 99-103)

11. FSAR Appendix A.5, Section A.5.1, Clarification of PBNP FSAR Requirements Regarding the Interface Between Seismic Class I to Lower Class System Boundaries.

The FSAR change clarifies the licensing basis requirements regarding the interface between seismic Class I and lower class system piping boundaries to be consistent with the PBNP design basis. Specifically, the change revises FSAR, Appendix A.5, Section A.5.1, "Definition of Seismic Design Classifications," to change the statement, "The interface between a Class I system and a lower Class system is at a normally closed valve or a valve which is capable of remote operation from the control room," to "The interface between a Class I system and a lower class system is at a normally closed valve, a valve which is capable of remote operation from the control room, or a valve which is capable of self actuation."

Summary of Safety Evaluation: The interface criteria in FSAR Appendix A.5 allows for the use of normally open valves at the seismic boundary interface, if they are capable of remote operation from the control room. The change adds "a valve which is capable of self actuation" to the list of acceptable interface valves. Although the exact failure mechanisms vary from valve to valve, the malfunction of concern is the failure of the valve to shut to isolate the Class I piping from the lower class piping. The failure of a valve to change state to isolate is applicable to both normally open remote manual valves and normally open valves capable of self actuation. The use of valves capable of self actuation is a clarification that is consistent with original plant design and licensing basis, design philosophy, and current regulatory guidance. The use of a self actuated valve is believed to be consistent with Westinghouse design philosophy at the time PBNP was designated in that such a valve "can practicably be closed in time to maintain the necessary functions of the higher class component or system." The use of normally open valves capable of self actuation is an equivalent, industry-accepted means of boundary isolation comparable to what is permitted by the seismic interface criteria in the current FSAR. In addition, testing or preventive maintenance programs help ensure that these self actuated valves can shut upon failure of the lower class piping.

Since a comparable level of boundary isolation is provided, there is no increase in the radiological consequences of an accident, event, or malfunction of equipment important to safety. The use of self actuated valves, instead of a valve capable of remote operation from the control room, to provide this seismic class boundary isolation capability does not create the possibility of an accident or event or the probability of a malfunction of equipment of a different type than previously evaluated in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-101)

12. FSAR Figures 4.2-1 and 4.2-1a, Vent Valve (RC-592) and Pipe Cap Added to PRT Containment Vent Flange.

The change to Westinghouse drawings 541F091 Sheet 2 and 541F445 Sheet 2 adds a normally closed vent valve (RC-592) and pipe cap downstream of the PRT containment vent flange on line 3/4"-RC-151R-3. It also adds valves 1RC-592 and 2RC-592 to the associated operations checklists as normally shut and capped.

Summary of Safety Evaluation: The vent valve (and cap) appear to have been installed to eliminate the need to disassemble and make up the flange connection at the top of the PRT each time a containment vent path was required for the tank. The only function of the additional piping, vent valve, and cap is to provide a pressure boundary for the PRT in lieu of a blind flange that was originally shown on these drawings to provide this function. Calculations 97-0254 and 99-001 seismically evaluated the additional valves and piping and found them to be code compliant and operable. The valve design is the same as other valves connected to the PRT and associated piping. The additional normally shut vent valve and pipe cap serve as a passive pressure boundary in lieu of the blank flange that was previously shown on the drawings. The change does not pose a USQ nor does it require a change to the TS. (SE 99-042)

13. FSAR Figures 4.2-1, 4.2-2, Reactor Coolant System.

During installation of MR 97-113 on 2HX-1A, it was discovered that the lines 2" upstream of valves 2RC-545C and 2RC-547C on the stream generator channel vent lines increase in size from 3/4" NPS to 1" NPS. The Westinghouse drawings show these lines to be 3/4"-RC-2501R-14 piping over its entire lengths. Valves

2RC-545A&B and 2RC-547A&B are 3/4" isolation valves, so the pipe must change back to 3/4" NPS before reaching these valves. Drawing change notices change Westinghouse drawings 541F445 Sheet 1 and Sheet 2 to reflect the as-built condition of the plant. The drawings are also FSAR Figures 4.2-1 and 4.2-2.

Summary of Safety Evaluation: Based on thickness readings taken of the exposed section of 1" piping, it was discovered that the 1" pipe is built to the design standards outlined in DG-M02 for Westinghouse primary piping. Thickness readings of the exposed 1" NPS pipe verified its wall thickness to be 0.25", which corresponds to a Schedule 160 pipe. This is the rated size for Westinghouse Class 2501 piping smaller than 3". Calculation 99-0025 verifies that the 1" piping meets FSAR acceptance criteria. Therefore, there is no increase in the probability of accidents or malfunctions as described in the CLB. There is no reduction in the margin of safety of TS. In addition, since the only change involves a pipe diameter increase, there is no possibility for the creation of an accident or malfunction different than those evaluated in the CLB. Finally, since the primary piping from the steam generator channel head is 3/4" NPS, flow out of the primary system is limited. Therefore, there is no increase in radiological consequences of accidents evaluated in the CLB, such as a small break LOCA. The change does not pose a USQ nor does it require a change to the TS. (SE 99-026)

14. FSAR Figures 4.2-1 Sheet 2 and 5.2-34a, Valve Position Change of 1RC-538 and 1RC-539.

    The change to Westinghouse drawing 541F091 Sheet 2 shows valves 1RC-538 and 1RC-539 as shut during normal operation. FSAR Figures 4.2-1 Sheet 2 and 5.2-34a are changed as well. These valves are the PRT gas analyzer isolation valves that also serve as containment isolation valves. They are normally maintained shut to provide isolation of the PRT gas space sample line to the waste gas system gas analyzer. The actual valve position is not changed; only the position as indicated on drawings.

    Summary of Safety Evaluation: The drawing changes are passive in nature. The valves are currently maintained in the shut position, which is the position required to satisfy the safety function of the valves to shut for containment isolation. Changing the indicated normal valve position for RC-538 and RC-539 has no affect on the probability of occurrence of accidents in the CLB. The valves isolate the PRT gas space sample line to the waste gas system gas analyzer and serve as containment isolation valves. Failures of these valves are not considered an initiating event or precursor to accidents. The safety function of these valves is to automatically shut to limit the release of radioactivity to the environment through the RCS piping that penetrates containment. No changes are made that affect the containment isolation function of the valves. The change does not pose a USQ nor does it require a change to the TS. (SE 99-041)

15. FSAR Figures 5.2-14c and 6.2-1, Revision to Westinghouse drawing 110E035, Sheet 1.

    The change to Westinghouse drawing 110E035, Sheet 1 ensures it reflects the normal status of valve 2SI-846, accumulator nitrogen supply AOV. The valve was shown in the normally open position on the drawing. The valve should be shown as normally shut. This correction is also made to FSAR Figure 6.2-1 Sheet 1.

    Summary of Safety Evaluation: The safety function of 2SI-846 is to shut on a containment isolation signal to ensure containment penetration 14c is sealed. Since the flow-path is one normally not used, except when nitrogen is required, 2SI-846 is kept normally shut. This ensures the safety function of the valve is met. The drawing change does not affect the way the SI system responds to an accident or malfunction. Reflecting the proper position on drawings enhances plant status control. Maintaining the valve in a normally shut position as required by approved plant procedures removes a possible failure mode (failing to shut) from unnecessarily affecting containment integrity. The change does not pose a USQ nor does it require a change to the TS. (SE 99-120)

16. FSAR Figure 5.2-32c and 9.3-2, Revision to Westinghouse drawings 684J741, Sheet 3 and 685J175, Sheet 3.

    The change to Westinghouse drawings 684J741 Sheet 3 and 685J175 Sheet 3 applies to a note related to CV-1296. The note is to read: Special valve, functions as both an isolation valve and relief valve." The change is in support of flow scanner diagnostic testing (under atmospheric conditions) that showed the valve opened at a calculated differential pressure of 1325 psid and not the 200 psid as previously stated on the drawings.

Summary of Safety Evaluation: CV-1296 originally was to provide overpressure protection for the charging pumps before the relief valves associated with charging pumps would relieve. During normal operations, CV-1296 is isolated from the charging pumps discharge header and the charging pumps relief valves provide the overpressure protection for the charging pump discharge piping. In response to GL 96-06, PBNP credited CV-1296 for providing the overpressure protection for penetration P-32C. The response also noted that the evaluated differential to lift the valve would be 1100 psid, which the piping stresses were evaluated to be below piping code allowables; however, during flow scanner diagnostic testing the valve was calculated to relieve at 1325 psid. The calculation did not account for stem rejection load; however, Calculation 99-0016 determined that the increased stresses on the piping were less than the piping code allowables and therefore would not change the response to GL 96-06. Therefore, there is no increased risk of causing an accident or affecting equipment important to safety than that already analyzed. Calculation 99-0062 accounted for the stem rejection load. It evaluated that the valve had lifted prior to 1325 psid during testing. The calculation also determined the maximum air pressure to be used during diagnostic testing for operability concerns and added a safety margin to make it even more conservative.

The calculations demonstrate that the piping penetration would withstand a larger internal pressure because of the heatup of the trapped fluid in the section of piping. The calculations also show that valve CV-1296 would perform its function as a relief valve to protect the penetration at the larger differential pressure without increasing the risk of accidents or causing a malfunction to other equipment important to safety. The change does not pose a USQ nor does it require a change to the TS. (SE 99-059)

17. FSAR Figures 5.5-3, 5.5-11, 4.1.3-3, 4.1.3-8, Revision to Wisconsin Electric drawing PBC-218.

    The SE addresses changes to the FSAR to accurately reflect plant fire door information. The fire barrier segments in which these fire doors are installed are industrial fire barriers; consequently, the information changed on the FPER figures does not affect the Appendix R safe shutdown analysis.

    Summary of Safety Evaluation: These fire doors are not credited with the mitigation of Appendix R fires. The drawing changes do not affect plant safety, as described in the CLB. The drawing revision ensures that the doors are accurately reflected in the CLB and on the parent drawings. Additionally, this change clarifies our compliance with the requirements of Nuclear Electric Insurance Limited requirements for these fire doors. The change does not pose a USQ nor does it require a change to the TS. (SE 99-081)

18. FSAR Figure 6.2-1, Revision to Westinghouse 110E017, Sheet 1.

    The change to Westinghouse 110E017 Sheet 1 includes a change in valve type for 1SI-D-02 from a gate to globe valve. The change is reflective of that in the field.

    Summary of Safety Evaluation: No physical changes are made to valve 1SI-D-02, P-15A&B SI pump discharge drain. The valve serves as a drain isolation valve for the high head SI pump. The normally shut valve remains fully capable of performing its isolation function. The valve is not in the SI flow path and is not required to change position during accidents described in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-121)

19. FSAR Figure 9.1-2, Revision to Westinghouse drawing 110E018, Sheet 3.

    The change to Westinghouse drawing 110E018, Sheet 3 consists of indicating the component cooling system valves associated with the abandoned waste evaporator as shut.

    Summary of Safety Evaluation: The equipment is not used and its permanent abandonment is in progress. The waste evaporator has no current function for plant operations, accident mitigation, nor is its operation within the scope of the CLB. Additionally, the change removes some piping from the active component cooling loop for Unit 1 by shutting the valves and isolating a portion of piping. The effect of thermal overpressurization of the piping thus isolated was addressed through the original design with a relief valve (1CC-743C) installed to protect against this possibility. Since the component cooling water flow has been isolated to the waste evaporator per approved procedures, and the FSAR has been revised to indicate that component cooling for this equipment is no longer required, this change is administrative.

The waste evaporator is not relied upon to provide safety or accident mitigation functions as indicated by FSAR Table A.6-1, "The Waste Disposal System Serves no Emergency Function." Further, the isolation of the component cooling water to the waste evaporator reduces the attached piping and therefore the probability of failures in the component cooling system. The change does not pose a USQ nor does it require a change to the TS. (SE 99-086)

20. FSAR Figure 9.3-1 and 9.3-2.

The revision modifies the pipe configuration for the seal water filter vent lines, seal water return lines, and a non-regenerative heat exchanger test connection as shown in FSAR Figures 9.3-1 and 9.3-2. The figures show the configuration as a closed isolation valve and a downstream flange. The plant's current configuration is a closed isolation valve and a tapped downstream flange connected to a threaded male swagelok adapter and plug. The test and vent lines are typically in a closed position and are only utilized during maintenance operations and for restoring the chemical and volume control system (CVCS) to operation. This configuration change will be reflected on changes made to Figures 9.3-1 and 9.3-2 in FSAR Section 9.3.

Summary of Safety Evaluation: The change to a tapped flange and threaded swagelok fitting and plug is equivalent to the currently shown flange connection. The piping configuration will continue to maintain the integrity of the CVCS pressure boundary since the flanges are rated for system design pressure and the swagelok fittings are rated for a pressure greater than system design pressure. The change will only effect non-safety related and non-ASME Section XI piping. The swagelok fitting and plug will help to ensure that an unwarranted reactor coolant leak be minimized by maintaining an equivalent boundary to that of the flanged connection. The change will continue to reduce the risk of an uncontrolled leak from the CVCS.

The configuration change will continue to meet the original design intent by providing an equivalent boundary to that of a blind flange connection. The swagelok fitting and plugs along with the flanges, will help to maintain the CVCS pressure boundary and ensure that an uncontrolled leak be minimized. The change does not pose a USQ nor does it require a change to the TS. (SE 99-035)

21. FSAR Figure 9.3-2, Revision to Westinghouse drawing 684J741, Sheets 2 and 3.

The change to Westinghouse drawings 684J741 Sheets 2 and 3 relate to CV-D-11, excess letdown heat exchanger outlet drain valve and a change in valve type, from gate to globe valve.

Summary of Safety Evaluation: The drain isolation valve for the excess letdown heat exchanger is normally shut and not in the normal flowpath of excess letdown when in operation. Changing the isolation valve from a gate valve to a globe valve has no affect on the operation of the excess letdown heat exchanger or other equipment important to safety. Since the valve still performs an isolation function, it does not increase the probability of accidents or radiological consequences previously analyzed in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-117)

22. FSAR Figure 9.11-1, Revision to Westinghouse drawings 541F092, Sheet 1 and 541F448.

The change to Westinghouse drawings 541F092 Sheet 1 and 541F448 changes valve SC-970 from a normally open to a normally shut globe valve. The valve is left in the normally shut position per approved procedures. The change also makes a valve type change for SC-959 from a gate to a globe valve. The valve is a sample isolation valve. The change also adds position indicators and solenoid valves to air-operated valves SC-951, SC-953, SC-955, SC-959, and SC-966A-C.

Summary of Safety Evaluation: SC-970 isolates the demineralizers and the charging pump discharge from the VCT and the drain header. Flow through the line is normally isolated with SC-983, SC-988 and SC-989. Changing the normal position of the valve has no affect on the primary sampling system operation, nor does it increase the likelihood of a failure of any part of the primary sampling system. Changing 1SC-959 from a gate to a globe valve does not reduce the ability of the valve to perform its isolation function, nor does it increase the likelihood of a failure of the valve. The components are currently installed and the change does not change the operation of the isolation valves. Parts of the primary sampling system are used post-accident to perform the required analyses as needed. System operation is controlled by approved plant procedures. Valve SC-970 is not manipulated in this situation. 1SC-959 is capable of allowing sampling of the Unit 1 RHR system as needed, since the change does not affect its operation. The change does not reduce the valves ability to isolate. Adding position indicators and solenoids to the drawings does not affect the operation of the sampling system isolation valves. They are isolated and can be opened when needed. Sampling requirements outlined in the TS are not affected. The changes do not reduce the margin of safety defined in the basis of TS. The change does not pose a USQ nor does it require a change to the TS. (SE 99-100)

23. FSAR Figure 10.1-1A, Revision to Bechtel drawing M-2201, Sheet 2.

The change to Bechtel drawing M-2201, Sheet 2 reflects existing test connections installed on the main steam crossover lines. No physical change is made to the system.

Summary of Safety Evaluation: The change has no affect on administrative controls or activities associated with the main steam system. The change does not impose restrictions on system availability or administration of setpoint activities.

The change does not affect the structural integrity of supports nor the system capability. The design function of the system remains unchanged. The change does not affect the operation, function, or method of performing a function of a SSC as described in the CLB, including the interim conditions. The change does not pose a USQ nor does it require a change to the TS. (SE 99-095)

24. FSAR Figure 10.1-6A, Revision to Bechtel drawings M-212, Sheet 1 and M-2212.

Valves 1&2AR-39A and 1&2AR-40A, Z-53A&B priming air ejector pressure indicator drain valves, are not illustrated within the FSAR.

Summary of Safety Evaluation: The change reflects material condition in the plant on Bechtel drawings M-212 and M-2212. The function of the priming air ejectors and associated equipment is unaffected by the change.

The change has no affect on administrative controls or activities associated with the priming air ejectors or their instrumentation. The change does not impose restrictions on system availability or administration of setpoint activities. The drawing change does not change procedures described in the CLB.

The drawing change does not affect the operation of the priming air ejectors nor its instrumentation. Neither the structural integrity of supports nor the system capability is effected. The change does not affect the operation, function, or method of performing a function of a SSC as described in the CLB, including the interim conditions. The change does not pose a USQ nor does it require a change to the TS. (SE 99-127)

25. FSAR Figures 10.2-1 Sheets 1 and 2, Revision to Bechtel drawings M-217 Sheet 1, and M-217 Sheet 2.

The drawing changes show that the inlet lines to the chemical add tanks tie into the 3" motor-driven auxiliary feedwater pump discharge piping (DB-3) prior to (upstream) that of its cross-tie line, and after (downstream) that of the valves AF-4012 and AF-4019. The change also adds locked open valves AF-20A and AF-59A. A locked open designation was also provided for valves AF-20A and AF-59A.

Summary of Safety Evaluation: AF-20A and AF-59A are located on the discharge pressure instrument lines for 1&2P-029, the turbine-driven auxiliary feedwater pumps (AFPs). A plant walkdown verified that these valves are locked open but not properly shown on the drawing. The auxiliary feedwater instrument valves are locked open to ensure proper turbine-driven AFP discharge pressure indication is provided to the control room in accordance with Regulatory Guide 1.97. The failure of these valves does not result in the loss of auxiliary feedwater system redundancy. The valves are designed to withstand a pressure higher than the design pressure of the turbine-driven AFP discharge piping, 1440 psig at 100°F (pipe Class DB-3). Therefore, the operation of the auxiliary feedwater system is not affected by the change. Failure of these valves does not result in a leak that can not be isolated.

The as-built location of the chemical add tank inlet lines does not change the piping design requirements for these lines or the motor-driven AFP discharge cross tie line. The stress analysis for the piping is not affected by the location of the chemical tank inlet connection to the motor-driven AFP discharge piping. The chemical add tanks are not equipment important to safety. The cross-tie line is also isolated during an auxiliary feedwater accident mitigation. Therefore, there are no changes to the auxiliary feedwater flow paths, response time or flow capability. Therefore, the auxiliary feedwater system maintains its ability to mitigate the consequences of an accident as designed. The changes have no adverse affect on the operation of the auxiliary feedwater system during normal accident conditions. The change does not pose a USQ nor does it require a change to the TS. (SE 99-073)

26. FSAR Figure 11.1-1, Sheet 1, Revision to Westinghouse drawing 684J971, Sheet 1.

Valve WL-1610, waste holdup tank drain to sump tank, is shown twice (once in the open position and once in the shut position) on Westinghouse drawing 684J971, Sheet 1. The valve should be noted for normal operation in the shut position. The change reflects actual plant configuration.

Summary of Safety Evaluation: The change does not change actual valve position, system function or operation. No procedure changes are required as a result of the change. Because there is no change in system function or operation, there is no change in how systems respond to an accident or malfunction as stated in the FSAR. Also, since operation of the waste liquid system is unchanged, there is no possibility of an accident or malfunction of safety equipment of a different type than analyzed in the FSAR. The change does not pose a USQ nor does it require a change to the TS. (SE 99-055)

27. FSAR Figure 11.1-1 Sheet 2, Revision to Westinghouse drawing 684J971, Sheet 1A.

The chemical drain tank and the laundry/hot shower tank are part of the waste liquid system. The tank provides a means to collect potentially contaminated liquids for processing prior to discharge. The tanks collect liquids from the Chemistry laboratory, laundry, and shower drains. The liquids are then pumped from the tanks to the waste holdup tank (WHUT) where it is stored until processing is done via the blowdown evaporator (BDE). Both tanks use a bubbler type level indication system. The drawing changes reflect actual plant configuration.

Summary of Safety Evaluation: The chemical drain and laundry/hot shower tanks and their piping and components are not safety-related nor are they important to safety. Accidents or events that might occur relative to this system have either been analyzed in the CLB or are bounded by accidents already evaluated in the CLB. FSAR Section 14.2.2 analyzes piping or tank ruptures in systems in the primary auxiliary building (PAB) that carry potentially radioactive liquids. The analysis states that the PAB sumps and basement can hold the contents of a single CVCS holdup tank. The combined volume of the chemical drain and laundry/hot shower tanks is about 1200 gallons, which is much less than the 58,400 gallons of a CVCS holdup tank. Therefore, a failure of the chemical drain and/or laundry/hot shower tanks would be contained within the PAB and no unmonitored release would occur. The change does not pose a USQ nor does it require a change to the TS. (SE 99-088)

28. FSAR Figure 11.1-1, Sheet 2, Revision to Westinghouse drawing 684J971, Sheet 1A.

The change to Westinghouse drawing 684J971, Sheet 1A updates valve 1WL-1721, P-18/P-66 reactor coolant drain tank, to a normally shut valve. Valve 1WL-1721A is also changed to show the valve as shut and capped.

Summary of Safety Evaluation: The waste liquid system collects and stores liquid wastes from plant primary systems. One collection point is the reactor coolant drain tank (RCDT). One RCDT in each containment collects liquid from various sources inside of its respective containment. Periodically the RCDT is pumped down using one of two RCDT pumps that are aligned in parallel outside of containment. Valves 1WL-1721 and 1WL-1721A are normally shut per approved plant procedures. The change improves the drawing to reflect the actual plant configuration. No changes are made to components or system operation. The change does not pose a USQ nor does it require a change to the TS. (SE 99-132)

29. FSAR Figure 11.1-2, Revision to Wisconsin Electric drawing PBM-225.

The change to Wisconsin Electric drawing PBM-225 corrects information for piping BEB-4-152 and BWL-9-152, and the addition of valve BE-00034A.

Summary of Safety Evaluation: FSAR Section 14.2.2 analyzes the rupture of various tanks that contain radioactive liquid wastes. The rupture of one of these tanks would be contained within the PAB, its basement, and its sumps. The contents of the tanks associated with the blowdown evaporator (BDE) system would flow to the floor of the BDE building. The floor drains of the BDE building are directed to the waste holdup tank via the sump tank and the sump tank pumps. The BDE and associated components are not safety-related nor are they important to safety. Accidents or events that might occur to this system or its components were analyzed in the CLB or are bounded by accidents already evaluated in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-032)

30. FSAR Figure 11.1-2, Revision to Wisconsin Electric drawing PBM-225.

The change to Wisconsin Electric drawing PBM-225 corrects valve positions for BE-35, HX-142 overhead condenser vent line control bypass valve, LSHL-LW-3, HX-140 BDE level switch, PI-LW-39, HX-144 BDE reboiler steam supply pressure indicator, and BE-64 P-34 BDE bottoms pump suction.

Summary of Safety Evaluation: The BDE system processes liquid wastes from the WHUT. The BDE distillate is then sent to a waste distillate tank for sampling and then released. The BDE system is described in FSAR Section 11.1. The BDE and radwaste systems and associated components are not safety-related nor are they important to safety. Changes pertain to the BDE and the radwaste systems and have no effects outside of these systems. The changes reflect current plant configurations and do not change component or system functions. FSAR Section 14.2.2 discusses the rupture of various tanks that contain radioactive liquid wastes. The rupture of one of these tanks would be contained within the PAB, its basement and its sumps. FSAR Section 14.2.3 discusses the rupture and release of the contents of one full gas decay tank or one VCT as a worst case scenario. It concludes that such an event would present no undue hazard to the public health and safety. The radwaste system is not discussed in the CLB. The liquid contents of the tanks or components associated with the BDE system would flow to the floor of the BDE building. The floor drains of the BDE building are directed to the waste holdup tank via the sump tank and the sump tank pumps. The gaseous contents of the tanks or components associated with the BDE system would be contained within the BDE building and vented to the PAB. PAB ventilation is filtered and monitored. The change does not pose a USQ nor does it require a change to the TS. (SE 99-111)

31. FSAR Figure 11.1-3, Revision to Westinghouse drawing 684J971, Sheet 2.

The change to Westinghouse drawing 684J971, Sheet 2 corrects actual valve positions incorrectly noted on that drawing.

Summary of Safety Evaluation: The BDE system and associated components are not safety-related nor are they important to safety. Accidents or events that may occur to this system or associated components were analyzed in the CLB or are bounded by accidents already evaluated in the CLB. FSAR Section 14.2.2 discusses the rupture of various tanks that contain radioactive liquid wastes. The rupture of one of these tanks would be contained within the PAB. The components, valves and piping involved with this change are located in the PAB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-110)

32. FSAR Figure 11.2-1, Revision to Westinghouse drawing 684J972, Sheet 1.

    The change to Westinghouse drawing 684J972, Sheet 1 adds WG-1027A, WG-1027 waste gas reuse header pressure control valve (PCV) sensing isolation to the diagram.

    Summary of Safety Evaluation: The waste gas system is mainly used as a cover gas system for various tanks. The system receives gas from these tanks when the tanks fill with liquid and compresses the gas for storage in the online gas decay tank (GDT). The online GDT also provides gas for reuse when the tanks are emptied of liquid. WG-1027 senses the pressure in the reuse header and opens or shuts to control that pressure. A sensing line from the PCV taps into the reuse header downstream of the PCV to help determine what corrections need to be made to the PCV position to maintain its setpoint. An isolation valve, WG-1027A, for this sensing line was added at some undetermined date without documentation. WG-1027A allows the PCV controller to be isolated from the reuse header. The change involves the waste gas reuse header sensing line isolation valve. The valve and the waste gas system are not safety-related. Accidents or events possibly related to this change are bounded by existing analyses found in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-030)

33. FSAR Figure 11.2-1 Sheet 3, Revision to Westinghouse drawing 684J972, Sheet 3.

    The change to Westinghouse drawing 684J972, Sheet 3 clarifies and updates the drawing to current plant configuration.

    Summary of Safety Evaluation: The change involves the gas analyzer, which is part of the waste gas system. The gas analyzer and the waste gas system are not safety-related. The only part of the waste gas system dealing with components on this drawing that are important to safety are the oxygen elements, analyzers, and indicators. These components notify personnel of oxygen levels in various tanks on the primary side that could create an explosive atmosphere. The change made to the oxygen sensing components involve swapping numbers of two channels on the drawing. This is editorial. Possible accidents or events related to drawing changes in the gas analyzer and the waste gas systems are bounded by existing analyses described in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-037)

34. FSAR Figure 11.2-2, Revision to Wisconsin Electric drawing PBM-226.

    The change to Wisconsin Electric drawing PBM-226 corrects the signal origin of the temperature switch high/low, annotates that TCV-GW9A is a fail open valve vice a fail shut valve and adds information as to the origin of various chart recorder inputs.

    Summary of Safety Evaluation: The temperature switch high/low receives its input from the E/I converter that converts the voltage signal from the temperature element to a current signal. Since current is proportional to voltage, the temperature switch high/low functions the same as previously illustrated; however, the magnitude of the signal changes. Therefore, there is no change to the operation or function of the equipment, just a correction as to the origin of the signal, which complies with the signal flowpath of the electrical drawings for the letdown gas stripper. TCV-GW9A is a temperature control valve that controls the cooling medium (radwaste component cooling water) to the heat exchanger. Since the letdown gas stripper package is not safety-related, during an accident condition, the radwaste component cooling water system is isolated from the component cooling water system upon receipt of a containment isolation signal. Changing the valve to fail open ensures that the cooling medium still flows through the heat exchanger, thereby preventing a failure of the heat exchanger or the radwaste component cooling water system. The change does not pose a USQ nor does it require a change to the TS. (SE 99-128)

35. FSAR Section 5.1.2.6 and 8.0.1.

FSAR text contained in Sections 5.1.2.6 and 8.0.1 is deleted to remove references to marinite board fire barriers previously intended to separate cable trays carrying redundant safety-related cabling. The subject barriers are constructed of Marinite-36, Type A, plain finish asbestos fiberboard. The FSAR sections refer to these barriers as "fire barriers;" however, the structures installed in the plant have no recognized fire resistance rating nor are they credited as fire barriers in the FPER.

Summary of Safety Evaluation: The original PBNP Fire Hazards Analysis, dated June 20, 1977, credited these barriers as preventing the propagation of a cable tray fire between redundant divisions of safety-related cable trays. The protection was considered to serve as a portion of the defense-in-depth concept of fire protection as discussed in BTP APCSB 9.5-1. The NRC FPSER and supplements to that document did not accept these barriers as providing sufficient separation of redundant divisions of cabling to consider them as a component that ensures the operability of safety-related cabling when an adjacent cable tray has been damaged by fire. PBNP eventually abandoned the attempt to credit these barriers as separating redundant divisions of cabling when it developed an alternate safe shutdown capability in the 1982-1983 time frame. Consequently the marinite board barriers have never been credited as a barrier that ensures fire safe shutdown of the plant.

The performance of these barriers is no longer credited nor necessary in the event of a fire. Post-fire safe shutdown of the plant is dependent on the availability of alternate shutdown equipment located in other areas of the plant to which the fire will not spread. Fire protective features guarantee assurance that fire will not spread to these other areas of the plant independent of the performance of the marinite board installations. Because the marinite board fire barriers serve no credited purpose, there is no need for their discussion in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-020)

36. FSAR Section 9.10, Fire Hose Hydrostatic Testing.

The change to the FSAR, FPER, and fire hose hydrostatic test procedures is consistent with NRC commitments that allow transition to 18-month operating cycles without having to perform hydrostatic testing of containment fire hose reels during power operation. Fire hose hydrostatic testing is accomplished via procedures TS 75, "Biennial Service Testing of Fire Hose and Fire Hose Station," and PC 73 Part 3, "Service Testing of Fire Hose and Fire Hose Stations." TS 75 performs biennial testing of fire hose reels whose operability is required by the Fire Protection Evaluation Report, while PC 73 Part 3 performs annual testing of containment hose reels and biennial testing of hose reels not identified in the FPER, including spare fire hoses. With the revisions, fire hoses are hydrostatically tested according to the following frequencies: External hoses (stored in hose houses) - annual versus biennial, internal hoses (except containment hose reels) - every 3 years versus biennial, and containment hose reels - refueling interval, nominally 18 months versus annual.

Summary of Safety Evaluation: Future testing will continue to be performed in accordance with NRC guidance (10 CFR 50 Appendix R Section III.E) and in accordance with NEIL insurance requirements. The hydrostatic tests ensure that the fire hoses remain capable of performing their function. During hydrostatic testing, the hoses are removed and tested to 250 psi using a hydrostatic pump. Test pressure is well in excess of the maximum fire protection system pressure of 175 psi, and maximum service water pressure of 125 psi. Since fire hoses are normally isolated from system pressure by a manual isolation valve, none of the plant fire hoses can initiate an accident or event previously evaluated in the CLB. The fire hose hydrostatic test frequencies are still performed at intervals sufficient for ensuring that the hoses are capable of performing their CLB functions (manual fire suppression, fire water supply to the condensate storage tanks or spent fuel pool). The change does not pose a USQ nor does it require a change to the TS. (SE 99-010)

37. FSAR Section 10.1 and Section 14.1.9.

A change to FSAR Section 10.1 relates to the detail of the opening time of the condenser steam dump valves. The FSAR used to state that the steam dump valves are designed to go from shut to the fully open position in less than 3 seconds. This is valve design specification information, and is not appropriate in the CLB.

Summary of Safety Evaluation: The condenser steam dump valves are non-safety related, non-QA, and non-EQ. The condenser steam dump "system" is designed to prevent a reactor trip following a 50% load rejection. In this event, the reactor power is reduced to a new equilibrium power level at a rate consistent with the capability of the rod control system. After 3 seconds, the condenser steam dump "system" is designed to pass 40% steam generator steam flow at full load. This is equivalent to 2,640,000 lbm/hr of steam at 821 psia, or 330,000 lbm/hr per valve. This is an important point because the design only models the condenser steam dump "system," not the individual steam dump valves. The FSAR description of the steam dump valves is changed to state the valves "rapidly" open. The valves rapidly open when individual solenoid valves apply instrument air at normal air pressure directly to the steam dump valve operators. The design specification and analysis assumption information are more appropriate for documents like the system DBD or the WCAPs.

Failure of a steam dump valve or the steam dump system within 3 seconds does not adversely affect the rod control system or the instrument air system. The steam dump system responds to plant transients and reduces challenges to engineered safety features like the steam generator safety valves and reactor over temperature ΔT trips. The reactor protection functions are not degraded and failure of the steam dump valves or the steam dump "system" is not affected. The rod control system and the instrument air system can still perform their intended function. The capability of the condenser steam dump "system" to perform as assumed in the transient analysis is not changed. The functionality of the condenser steam dump "system" is maintained by the performance of acceptance testing, post-maintenance testing, and periodic testing as required. Failure of the condenser steam dump "system" is bounded by a reactor trip and closure of the safety-related main steam isolation valves. The steam dump valves are not relied upon to mitigate the consequences of a radiological event and no TS action statements associated with the condenser steam dump valves exist.

The change to the FSAR does not change the design or method in which the steam dump "system" performs its function. FSAR Section 14.1.9 analyzes a complete loss of load and does not take credit for the steam dump system. However, by making this change valve design specification information that does not belong in the current licensing basis will be removed, the FSAR will be more accurate, and unnecessary equipment maintenance may be avoided. The change does not pose a USQ nor does it require a change to the TS. (SE 99-090)

38. FSAR Section 14.1.4, CVCS Malfunction.

The assumption in FSAR Section 14.1.4 for the number of charging pumps that affects the dilution rate is inaccurate. Through a flow measurement test, it was determined that FCV-111 can pass approximately 100 gpm when the valve is full open, three charging pumps idle, one reactor makeup water pump running, and the RCS is depressurized. The test simulated worst case conditions, with a high upstream pressure provided by the reactor makeup water pump running and the low downstream pressure with the RCS at atmospheric pressure. It was also determined that when no charging pump is in service, the pumps and associated piping offer no measurable flow resistance. With no charging pumps in service, the dilution rate is limited by the maximum flow through FCV-111 of approximately 100 gpm. FSAR 14.1.4 is revised to clarify the relationship between the number of charging pumps, the flow through FCV-111, and the rate of addition of unborated water.

Summary of Safety Evaluation: FSAR Section 14.1.4 states that unborated water can only be added when a charging pump is running. This is deleted because unborated water can be pushed through the charging pumps when all three pumps are idle during cold shutdown and refueling cases when RCS pressure is low. Not only do the number of charging pumps influence the rate at which unborated water can be introduced into the RCS, but the maximum flow capacity of FCV-111 also limits the dilution rate. Information presented in the general description only applicable to the power and startup dilution analyses is deleted because this level of detail (e.g., the limiting dilution flow) is already presented in each specific dilution analysis section.

The boron dilution during cold shutdown analysis is revised to accommodate no charging pumps in service. The FSAR change modifies Figure 14.1.4-1 such that the idle charging pump condition is addressed using the two pump curve (120 gpm). The third pump is required to be out of service per low temperature overpressurization (LTOP) requirements. This bounds all possible charging pump configurations. Also, the dilution flow associated with each charging pump configuration is added back into the figure per the original NRC submittal figure. The change does not pose a USQ nor does it require a change to the TS. (SE 99-034)

39. FSAR Section 14.2.4, Steam Generator Tube Rupture.

    The change adds information to the general description of FSAR Section 14.2.4. It presents the conclusion of a Westinghouse assessment regarding operator action time to terminate primary-to-secondary fluid leakage.

    Summary of Safety Evaluation: The assessment performed by Westinghouse as documented in WEP-98-055 approximates the impact of increasing the operator action time assumed for break flow termination by comparing the Point Beach methodology (which does not model operation actions) with the LOFTTR2 analysis methodology. The assessment and conclusions are based on the assumption that no steam generator overfill occurs. This assumption was validated during operator simulator training. The overall conclusion of the Westinghouse assessment is that the licensing basis SGTR analysis remains applicable and does not need to be revised, even though operator action time may exceed the assumed break termination time of 30 minutes. The conservatism in the methodology because of modeling operator actions enables the operator action time to be increased beyond 30 minutes without increasing the accident consequences. Break flow termination could be increased to approximately 44 minutes without exceeding the integrated break flow, ruptured steam generator atmospheric releases and radiological consequences in the current licensing basis SGTR analysis.

    The change does not increase the probability of occurrence of an accident or event, or increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the CLB. The change does not create new failure modes or new accident initiators. There is no increase in the radiological consequences of an accident. The change does not increase the probability of occurrence of an accident or event of a different type, or increase the probability of occurrence of a malfunction of equipment important to safety of a different type than previously evaluated in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-043)

40. NFPA 20 Deviations, FPER Section 4.2.1.1, C-061 Diesel Fire Pump Controller.

    Contrary to NFPA 20-1968 requirements, the C-061 diesel fire pump engine controller does not have a locked door with a break glass panel and does not have an alarm that sounds in the control room if the controller switch is turned to "off" or "manual." The diesel fire pump controller is not in conformance with commitments made to the NRC. Rather than modify the controller to meet the NFPA requirements, this evaluation documents the acceptability of these deviations from NFPA 20.

    The FPER is revised to identify that the diesel fire pump controller deviates from the above NFPA 20 requirements.

    Summary of Safety Evaluation: NFPA 20 requires diesel fire pump controllers to be located inside a locked cabinet with a break glass panel with remote annunciation in the event that the control switch is placed in the "off" or "manual" position in order to ensure that the organization responsible for the pump is assured of the ability to maintain the engine in a standby condition. The NFPA standard does not contain guidance for the accessibility of the room containing a diesel fire pump. Since it is possible for fire pumps to be installed in areas that are not well-controlled, NFPA standards have specified means of ensuring that the controller is not disabled without the owner being aware of the pump status. At PBNP, the diesel fire pump and its controller is contained in the south service water pump room. Individuals entering this room are trained on the importance of not manipulating operational plant equipment or are under the direct escort of someone who is cognizant of this requirement. Access to this area is controlled by a locked door with card reader access. Therefore, the possibility of someone intentionally disabling the diesel fire pump by moving the control switch out of the "auto" position is extremely remote. Additionally, there are many potential actions that may be taken, other than manipulating the main control switch inside C-061, that would disable operation of the diesel fire pump. Having an alarm on the control switch position does not provide a significant increase in the assurance that the fire pump is in its proper standby condition.

During diesel fire pump operation, the controls inside the diesel fire pump control panel are manipulated by trained operators working under procedural guidance that ensure that the control switch is not placed in the incorrect position. During monthly diesel fire pump testing, the control switch is not moved from its normal "auto" position. The pump is started by simulating a drop in fire system pressure and the pump is stopped by pressing the "reset" pushbutton inside the C-061 controller. Therefore, the possibility for inadvertent manipulation of the control switch is not significant and PBNP has an equivalent level of protection to that specified in NFPA 20. In addition, PBNP has had no instances of the diesel fire pump control switch identified in positions other than the "auto" standby status position. The change does not pose a USQ nor does it require a change to the TS. (SE 99-015)

41. Primary-to-Secondary Leak Rate Monitoring Program.

The SE describes the primary-to-secondary leak rate monitoring program. Point Beach procedures now reflect the new program. The primary-to-secondary leak rate monitoring program is patterned after EPRI TR-104788-R1 "PWR Primary-to-Secondary Leak Guideline-Revision 1," in which action levels are created. The primary-to-secondary leak rate monitoring program includes trending and monitoring requirements at normal operating conditions, increased monitoring conditions (leakage >5 gpd but <30 gpd), action Level 1 (leakage greater than or equal to 30 gpd but less than or equal to 150 gpd), and shutting down the plant at action Level 2 (leakage greater than or equal to 150 gpd or accelerated rate greater than or equal to 60 gpd/hr). The program also has contingencies to take if the unit specific air ejector is inoperable.

Summary of Safety Evaluation: The program is modeled after EPRI Primary-to-Secondary Leak Guidelines that studied several tube ruptures in industry. The program's maximum leak rate of 150 gpd for shutdown or increasing leak rate of 60 gpd/hr provides assurance against tube rupture at normal and faulted conditions and helps ensure that cracks that might grow at a much greater rate than expected are detected by leakage. The limit also helps ensure that the dose contribution from tube leakage remains less than 10 CFR 100 dose limits for postulated faulted events. The maximum limit is also less than the TS limit for shutdown of 500 gpd primary-to-secondary leak rate. The change does not pose a USQ nor does it require a change to the TS. (SE 99-116)

42. SPEED 99-037, Slings.

The change involves our commitment to ANSI B30.9, 1971 Edition. The change evaluated removes the specific reference to the 1971 Edition and changes it to a general reference to the "most recent evaluated Edition." The change is in support of current manufacturer's parts being processed in accordance with ANSI B30.9, 1990 Edition.

Summary of Safety Evaluation: As demonstrated by the change from the 1971 Edition to the 1990 Edition, code changes often do not involve actual changes in requirements; therefore, allowing for the use of the "most recent evaluated edition," and non-technical changes of the Code edition can be effected using the spare parts equivalency evaluation document (SPEED) process and a screening versus a full evaluation. Where more technical changes are involved, the restrictions of the scope of SPEED would drive the process to a more rigorous evaluation. A review of the rated capacity tables for natural and synthetic fiber rope slings in

ANSI B30.9-1971 and ANSI B30.9-1990 show no change in the rated capacities. A review of the synthetic webbing sling sections of both these editions, shows that the 1971 Edition did not contain rated capacity tables. The 1990 Edition has capacity tables, thereby making the standard more restrictive and decreasing the probability of an accident or event. The change does not pose a USQ nor does it require a change to the TS. (SE 99-045)

43. SPEED 99-048, Replacement of Thomas Pump with Gast Pump.

    The SE evaluates the acceptability of replacing the present OEM-supplied air radiation monitor diaphragm pumps with rocking piston pumps. SPEED 99-048 documents the equivalency of the P-707A air pumps. FSAR Section 11.5 states the special particulate iodine noble gas (SPING) monitors use sealed diaphragm air pumps. The air pumps presently installed in the radiation monitors are no longer manufactured; therefore, a replacement pump is necessary.

    Summary of Safety Evaluation: The replacement pump has equivalent airflow and electrical characteristics to that of the original design. The operation, function and airflow of the radiation monitors remain unchanged. The change does not increase the probability of occurrence of an accident previously evaluated in the FSAR. The radiation monitors have no accident initiation mechanisms. The change does not increase the probability of occurrence of a malfunction of equipment previously evaluated in the FSAR. The replacement pumps have a more advanced pumping mechanism which is as reliable as the diaphragm assembly in the original components. The change does not increase the consequences of an accident or malfunction of equipment. The radiation monitors have no nuclear safety-related mitigation function. The change does not create the possibility of an accident of a different type than previously evaluated in the FSAR. The subject radiation monitors do not have accident initiation mechanisms. The change does not create the possibility of an accident of a different type than previously evaluated in the FSAR. The subject radiation monitors do not have accident initiation mechanisms. The failures of the pump include stoppage and leakage. Flow alarms are permanently installed on the monitors to detect these conditions. Calculation 99-0049 states the minimum required flow for the containment air monitors to be 0.296 lpm. The replacement pumps are sized for 60 lpm. The change does not pose a USQ nor does it require a change to the TS. (SE 99-050)

44. Storage of Low-Level Dry Radioactive Waste, Green-Is-Clean Radioactive Waste, and Radioactive Material in PBNP RCA Yard Areas.

    Dry low-level radioactive waste, green-is-clean radioactive waste, and radioactive materials are temporarily stored in 20' and 40' SeaLand containers located in outside yard areas of the radiation controlled area (RCA). The materials are collected in the plant, bagged or wrapped, and transported to SeaLand containers for storage. The waste is stored until the SeaLand container is full, at which time it is shipped offsite to a radioactive waste processor. The SeaLand containers provide for economical processing of the waste. Typical storage time for waste materials has been 6-7 months during a 12-month reactor operating cycle and 2-3 weeks during refueling outages. Expected storage time for waste materials during an 18-month reactor operating cycle is 9-10 months.

    Summary of Safety Evaluation: FSAR sections describing the solid waste management system and radioactive material storage are changed to describe the temporary storage of dry low-level radioactive waste and radioactive materials in outside yard areas of the RCA. Storage of SeaLand containers or B25 boxes in RCA outside yard areas does not affect equipment important to safety because the storage location and storage conditions prevent the movement of the containers under flood and high wind conditions. Dose rates on contact with and at 2 meters from the surface of the container are controlled to meet shipping requirements. As a result, the probability of evaluated dose accidents, events, or equipment malfunctions remain unchanged. Additionally, there are no mechanisms whereby the possibility of new equipment malfunctions can be created.

    A dose calculation determined that the maximum organ or whole body dose is 0.2 mrem to a member of the general public at the site boundary which is well within the dose limits of 49 CFR 190 for a release involving a large amount of the contained waste (10% of the waste in the container which is equivalent to a waste weight of 640 lbs). Container integrity is detected by periodic inspections and restored well before a release of this quantity of waste is reached.

Administrative controls are established for the placement of SeaLand containers in RCA outside yard areas, for routine surveys and container integrity inspections, and actions to take upon detection of a failure of container integrity. The change does not pose a USQ nor does it require a change to the TS. (SE 99-007)

45. Unit 1 Cycle 26 (U1C26) Reload.

The SE addresses the U1C26 core loading pattern as described in Westinghouse Letter 99WE-G-0022. The core has 121 upgraded optimized fuel assemblies (OFAs) arranged in the core loading pattern. During U1R25, 40 upgraded OFAs are replaced with 16 fresh Region 28A (4.00 w/o U-235) and 24 fresh Region 28B (4.700 w/o U-235) upgraded OFAs. The SE covers the mechanical design, nuclear design, thermal-hydraulic design, power capability, and FSAR accidents that apply to the U1C26 reactor core and covers all modes of operation for U1C26. The reload core involves a potential change to the facility or its operation as described in the FSAR. The SE includes the U1C26 reload design and safety analyses performed by Westinghouse as part of their responsibility for reload fuel supply and core design and additional evaluations performed by WE. Westinghouse reported the results of the cycle design and safety analyses in the U1C26 final reload safety evaluation, dated October 15, 1999. The effect of the U1C26 design on the boron dilution event in cold shutdown is assessed in Calculation 99-0033.

Summary of Safety Evaluation: Westinghouse concluded that the following conditions are to be met during the U1C26 reload: The end-of-cycle 25 burnup is bounded by 15,100 to 16,785 MWD/MTU; the U1C26 burnup does not exceed the end-of-full power capability (defined as control rods fully withdrawn and less than or equal to 10 ppm of boric acid at the U1C26 rated power condition of 1518.5 MWt) plus up to 1,500 MWD/MTU of power coastdown operation; there is adherence to the plant operation limitations given in TS; the safety aspects on the reactor internals of utilizing PPSAs have been assumed by WE (the NRC approval of TSCR 127 included approval of the use of PPSAs); and the effect of the U1C26 design for boron dilution event in cold shutdown has been assessed by WE.

The U1C26 reload core design meets applicable design criteria or has been shown to maintain the same levels of safety as considered in the reference design basis evaluations and ensures that pertinent licensing basis acceptance criteria are met or are justified as acceptable. Though fuel and core design are not directly related to the probability of previously evaluated accidents or events, the demonstrated adherence to applicable standards and acceptance criteria precludes new challenges to components and systems. The containment is evaluated to maintain integrity during a steam line break (SLB) in Calculation N-89-042, by comparing PBNP to a similar 2-loop plant. The calculation is valid as long as the core does not return to critical following a SLB. Westinghouse has performed an evaluation which confirms that the U1C26 core will not return to critical following a SLB. Therefore, the evaluation of the containment integrity during a SLB is valid. No new performance requirements are being imposed on systems or components such that design criteria is exceeded nor do the changes cause the core to operate in excess of pertinent design basis operating limits. The change does not pose a USQ nor does it require a change to the TS. (SE 99-124)

46. Unit 2 Cycle 24 (U2C24) Reload.

The SE addresses the U2C24 core loading pattern as described in Westinghouse Letter 98WE-G-0067. The core has 121 upgraded OFAs arranged in the core loading pattern. During U2R23, 53 OFAs are replaced with 33 fresh Region 26A (4.600 w/o U-235), and 20 fresh Region 26B (4.900 w/o U-235) OFAs.

The SE covers the mechanical design, nuclear design, thermal-hydraulic design, power capability, and FSAR accidents that apply to the U2C24 reactor core. The SE includes the U2C24 reload design and safety analyses performed by Westinghouse as part of their responsibility for reload fuel supply and core design and additional evaluations performed by WE. Westinghouse reported the results of the cycle design and safety analyses in the reload safety evaluation for U2C24, dated December 31, 1998. WE reported the results of additional evaluations in Calculation 98-0173.

Summary of Safety Evaluation: Westinghouse concluded that the following conditions are to be met during U2C24: The end-of-cycle 23 burnup is bounded by 11,300 to 12,300 MWD/MTU; the U2C24 burnup does not exceed the end-of-full power capability (defined as control rods fully withdrawn and less than or equal to 10 ppm of boric acid at the U2C24 rated power condition of 1518.5 MWt) plus up to 1,500 MWD/MTU of power coastdown operation; there is adherence to the plant operating limitations given in TS; the safety aspects of utilizing PPSAs are assumed by WE; and, the effect of the U2C24 design for boron dilution event in cold shutdown has been assessed by WE.

The U2C24 reload core design meets applicable design criteria or has been shown to maintain the same levels of safety as considered in the reference design basis evaluations and ensures that pertinent licensing basis acceptance criteria are met or are justified as being acceptable. Though fuel and core design are not directly related to the probability of previously evaluated accidents or events, the demonstrated adherence to applicable standards and acceptance criteria precludes new challenges to components and systems. No new performance requirements are imposed on systems or components such that design criteria are exceeded nor do the changes cause the core to operate in excess of pertinent design basis operating limits. The change does not pose a USQ nor does it require a change to the TS. (SE 99-009)

Summary of Safety Evaluation: The SE revision addresses heatup above 200°F, reactor startup, and power operation. The SE revision addresses the reactor going critical following a steam line break inside containment accident.

The containment is evaluated to maintain integrity during a steam line break (SLB) in Calculation N-89-042, Revision 3, by comparing PBNP to a very similar 2-loop plant. The calculation is valid as long as the core does not return to critical following a SLB. Westinghouse Calculation CN-WEPCO-006 confirms that the U2C24 core will not return to critical following a SLB. Therefore, the evaluation of the containment integrity during a SLB is valid. No new performance requirements are imposed on systems or components such that a design criteria is exceeded nor do the changes cause the core to operate in excess of pertinent design basis operating limits. The change does not pose a USQ nor does it require a change to the TS. (SE 99-009-01)

47. Unit 2 Operation at Less Than 55% Power with Nos. 4A and 4B Feedwater Heaters Inoperable.

The SE evaluates reducing Unit 2 power to <55% and removes feedwater heaters Nos. 4A and 4B from service by isolating the shell side of the heaters. Feedwater flow through the tube side of the heater is maintained. This allows the feedwater heaters to be inspected for erosion of the heater shells or other maintenance activities. The condition could exist until the next refueling, U2R24. The 55% power level was selected based on conservative application of instructions provided in Westinghouse L.I. 1250-7521, "Feedwater Heater Operation." The required power reduction ensures that the increase in the extraction steam flow to the No. 5 heaters does not exceed full power extraction steam flow. Further assurance of the impact on plant response was provided by an engineering review of PEPSE secondary plant thermal performance analyses at 100% power at a higher power level in this condition (Nos. 4A and 4B heaters steam isolated) require the evaluation to be revised or a new evaluation to be performed. The evaluation does not address the activity to return the feedwater heaters Nos. 4A and 4B to service, or returning the unit to >55% power if the feedwater heaters are to remain inoperable.

Summary of Safety Evaluation: Operating with the steam flow to the Unit 2 Nos. 4A and 4B feedwater heaters isolated results in a lower feedwater temperature. The lower feedwater temperature is a concern from a thermal stress perspective, but this is bounded by conditions that exist during initiation of auxiliary feedwater flow (ambient temperature water). Further, this is a steady state condition after the down power and heat exchanger isolation is completed rather than a thermal transient associated with auxiliary feedwater initiation and testing.

The reduction in feedwater enthalpy accident analysis is bounded by the excessive load increase incident because it is less severe than the excessive load increase incident. Both the reduction in feedwater enthalpy and excessive load increase incident are less severe at lower power levels. Additionally, removal of the No. 4A and

4B feedwater heaters results in a smaller temperature change for the reduction in feedwater enthalpy case. The activity that operates Unit 2 with the Nos. 4A and 4B feedwater heaters shell side isolated while at power level of <55% does not involve a change to the TS. The change does not pose a USQ nor does it require a change to the TS. (SE 99-038)

48. WO 9709241, Addition of Pressure Indicators and Regulators to Gas Analyzer Sample Lines from the Train A and B CVCS Holdup Tanks.

    The gas analyzer system samples the oxygen and hydrogen concentration from 13 different sample lines. Eleven of the sample lines contain pressure regulators and pressure gauges. The SE addresses the addition of a pressure regulator and pressure gauge in the sample lines from the "A" and "B" CVCS holdup tanks to the gas analyzer (Z-46 panel). The addition of the regulator and gauge reduces the pressure coming from the CVCS holdup tanks to prevent possible over pressure damage to the gas analyzer system sample pump. The change makes the last two inputs to the gas analyzer system similar to the other gas analyzer inputs that currently have pressure regulators and pressure gauges on their sample lines.

    Summary of Safety Evaluation: The gas analyzer and the lines from the CVCS holdup tank are not safety-related and are not important to safety since they are not directly or indirectly credited in the accident analyses in the CLB. FSAR Section 14.2.3 discusses a rupture of the gas decay tank as the limit of the hazard that could result from malfunctions in the radioactive waste disposal system piping or components. The addition of the pressure regulator and pressure gauge does not change accidents or events in the CLB and are bounded by existing analyses of the waste gas system.

    The gas analyzer and CVCS sample lines are not safety-related and are not important to safety. Accidents or events related to the gas analyzer and sample lines are bounded by existing analyses contained in the CLB. The change does not pose a USQ nor does it require a change to the TS. (SE 99-094)

49. WOs 9818089 and 9818090, In-Situ Ultrasonic Test (UT) of Structural Lid-to-Shell Weld, WMSB-01 and WMSB-03, at the ISFSI.

    The SE evaluates an ultrasonic test (UT) of the structural lid-to-shell weld for two loaded multi-sealed baskets (MSBs) on the ISFSI pad (MSB-01 and MSB-03). The work is performed via WOs 9818089 and 9818090. The UT is in conformance with an NRC commitment to conduct a UT of the weld on these two casks by August 31, 1999.

    10 CFR 72.48 Evaluation Summary: The UT is performed by removing the weather cover on the ventilated concrete casks (VCCs) surrounding the MSBs, lifting the shield ring slightly with a special lift device in order to access the weld, scanning the weld, and replacing the weather cover on the VCC. Numerous design and administrative controls are employed in order to keep the VSC functional during the exam and to minimize occupational exposure to radiation and environmental effects. Aspects of the UT affect only SSCs licensed under 10 CFR 72, and had no effect upon SSCs licensed under 10 CFR 50. Therefore, they were not evaluated against 10 CFR 50.59. The administrative and design methods employed in this examination were such that, due to the non-intrusive nature of the UT exam, and the conditions established during the exam, the activity was deemed to be able to be made without requesting specific permission from the NRC. Although occupational exposure was increased, it was not deemed to be significant. Similarly, although an argument could be made for an increase in dose to the general public, the argument put forth demonstrated the insignificance of this increase, and therefore no decrease in the margin existing to the bounding regulatory limits. No USQ, significant increases in occupational exposure, significant unreviewed environmental impact, nor changes in the license conditions of the C of C are created. (SE 99-056)

50. WOs 9905802 and 9917996, Temporary Power for MCCs 1B-40, 1B-42, and B-43.

The change supplies temporary power to de-energized motor control centers (MCCs) 1B-40, 1B-42, and 1B-43. 1B-40 receives temporary power from the non-safety related portion of 2B-40. The temporary power is supplied via a temporary cable from 2B52-406H to 1B00-401M. 1B-42 receives temporary power from 1B-31. The temporary power is supplied via a temporary cable from B52-335C to B52-434BR. Loading on MCCs 2B-40, 1B-31, and B-33 have been checked to ensure that the temporary load does not exceed the load rating on each MCC.

Summary of Safety Evaluation: The normal supply to 1B-40 is shown on FSAR Figures 8.4-1, and 8-7. The normal supply to 1B-42 and B-43 is shown on FSAR Figures 8-1 and 8-8. The work orders reflect a deviation from those figures by supplying temporary power to 1B-40, 1B-42, and B-43 from 2B-40, 1B-31, and B-33. The change is for a short duration of less than 7 days, thus not necessitating a change to the FSAR. 1B-40 is taken inoperable and is supplied by the non-safety related portion of 2B-40. The temporary non-qualified, non-safety related supply breaker on 2B-40 is stripped on a loss of voltage signal. There are no train separation issues since 2B-40 and 1B-40 are both safeguards Train B and the cable is routed entirely within the EDG building. Compensatory measures are setup to address Appendix R concerns. 1B-31 and B-33 are stripped on an SI signal. The load breakers on 1B-31 and B-33 are sized to provide adequate protection to the supplied MCCs should a fault occur. The temporary cables that connect 1B-31 and B-33 to 1B-42 and B-43, respectively, are adequately sized to provide the MCCs with their requested load currents. The supply breakers are sized to protect the cable from malfunction or damage.

The activity ensures that heating and lighting are provided to the EDG building during early winter conditions, adequate lighting, SFP cooling, boric acid loads, and RHR loads are provided to the PAB, and adequate power is supplied to the RCDT pump, boric acid tank heater, and waste evaporator feed pump during the 1A-06 outage. The change does not pose a USQ nor does it require a change to the TS. (SE 99-123)

51. WO 9911668, (Unit 1 and 2), Auxiliary Feedwater.

MRs 97-129*A and 97-129*B replaced the 1&2AF-100 auxiliary feedwater first-off check valves. However, after the valves were replaced, leakage was identified on 2AF-100. It was determined that these valves were not properly seating. In order to get the first-off check valves to seat properly, the upstream pressure must be relieved. WO 9911668 installs a temporary modification that consists of needle valves and pressure gauges on the drain connections of the 2P-29 discharge lines. The work plan for the temporary modification directs operations personnel to install the pressure gauges, needle valves, and tubing at the drain connections. Then they open the drain valves, record the pressure readings, and carefully bleed the pressure out of the line using the needle valves and record pressures again. When this is complete, the drain valves are shut, and the valves, gauges and fittings remain installed. The temporary modification is removed when enough pressure data is collected.

Summary of Safety Evaluation: Several of the drain valves on the auxiliary feedwater lines are part of a system required to maintain containment integrity. Opening the drain valves upstream of the manual containment isolation valves (1AF-18, 1AF-19, 1AF-31, 1AF-44, 2AF-32, 2AF-45, 2AF-56, 2AF-57) represent opening part of a containment penetration boundary, but not a loss of containment integrity. A dedicated operator is stationed to shut these valves as directed by the control room. Other work orders and temporary modifications may be initiated in the future to install other pressure gauges and/or needle valves and to open the drain valves to record the auxiliary feedwater discharge line pressures and relieve excess pressure as needed to troubleshoot the check valve problem.

Some of the auxiliary feedwater drain valves that are opened are part of a system required to maintain containment integrity. However, opening the valves does not constitute a loss of containment integrity. These essential penetrations (P-05 and P-06) are required to be open post-LOCA. Steam generator isolation during a tube rupture requires manual action to shut the containment isolation valves. Also, loss of containment integrity does not constitute an accident or event. The minor amount of water leakage that would occur past the needle valve if the check valves do not seat does not increase the likelihood of a loss of normal feedwater (LONF) accident or cause a loss of the auxiliary feedwater pump. The small amount of water also does not represent a potential flooding concern. Therefore, this activity does not increase the probability of occurrence of an accident or event as evaluated in the CLB or of a different type than any previously evaluated in the CLB. If the turbine-driven auxiliary feedwater pump starts while the drain valves and a needle valve are open, water is forced out the opening; but this does not cause a malfunction of the pump because of a 60 gpm flow margin to the steam generators.

Failure of the fittings up to the needle valve is not likely since it is verified that components are appropriately rated to withstand pump discharge pressure. The additional weight of the installed components does not increase the likelihood of piping failure in a seismic event. No other type of malfunction can occur to the auxiliary feedwater pump during the evolution. If an accident occurred while the drain valves were open, the drain lines would not provide a flow path out of containment, since the auxiliary feedwater pump discharge lines are connected to a closed system inside containment, and these essential penetrations are required to be open post-LOCA. They do not perform a containment isolation function in this event. Steam generator isolation during a tube rupture requires manual action to shut the containment isolation valves per AOP-3. Also, the main and auxiliary feedwater systems are not a source of significant radioactivity. Therefore, the activity does not increase the radiological consequences of an accident, event, or malfunction of equipment important to safety as evaluated in the CLB. The turbine-driven auxiliary feedwater pumps are able to provide the required flow rate to the steam generators if necessary during the activity. The modification does not pose a USQ nor does it require a change to the TS. (SE 99-102)

52. WVSC-24-04, Thinner Weld Thickness, MSB Outer Valve Cover Plates.

The change involves using a smaller size for the outer valve cover plate weld than described in the ISFSI SAR (3/16" versus 1/4"). The affected cask was analyzed for the consequences of thinner weld material. The effects were previously analyzed in an operability determination for CR 98-3637, but were re-examined here to determine if there was an unreviewed safety question. The procedures were changed so future welds do not result in a thinner weld thickness.

Summary of Safety Evaluation: The examination concluded that the effect of the thinner weld material does not result in weld failure for any analyzed accidents. An extra accident scenario, the vertical drop, was examined in this report, which the manufacturer felt was not a probable accident scenario. It was examined here only to provide complete analysis of ISFSI Chapter 11.2.3 accident scenarios. No USQ, significant increases in occupational exposure, significant unreviewed environmental impact, nor changes in the license conditions of the C of C are created. (SE 99-016)

53. WVSC-24-04, Gasket FME.

While loading multi-assembly sealed basket (MSB) #4 (WMSB-04) via WO 9609951, a nitrogen purge and vent was established to control hydrogen concentration in the space beneath the shield lid (after WMSB-04 was loaded with fuel and moved to the decontamination facility). While executing this line-up, an auxiliary operator noticed a gasket missing from the female end of a cam-lock quick connect fitting. The female end of the cam-lock quick connect fitting is used to join two hoses within the decontamination facility that supply nitrogen to the MSB. The hose with the female cam-lock fitting was last used during the loading process to fill the MSB with borated water. There is a chance that the gasket was flushed into the MSB prior to loading it with fuel. The gasket was not verified present prior to filling the MSB with borated water. Because of this, a conservative assumption was made that the gasket is inside of WMSB-04 and evaluation performed declaring the container operable but degraded.

10 CFR 72.48 Evaluation Summary: The gasket was shown through analysis that is does not contain corrosive contaminants and does not float. There is no real transport mechanism that could block the vent at the shield lid during cask unloading operations. If the entire gasket would decompose, the 15 liters of methane produced represents only 0.3% of the total volume in the MSB. The 0.3% concentration of methane in the MSB is 1/16th the lower flammability limit of methane. We are currently monitoring for hydrogen and also utilizing a purge and vent process during unloading operations as a result of our commitments associated with NRC

Bulletin 96-04. By monitoring for hydrogen gas and by utilizing the purge and vent process, we ensure that a flammable atmosphere is not created within the MSB. Due to the reducing nature of methane, it may react with the cladding oxide layer on the fuel assemblies, but will not react with the zircalloy cladding or cask internals. This would not be detrimental to the cladding or retreivability of the fuel assemblies and MSB. During the loading process of the MSB, there was no noticeable reduction in the amount of time it took to draindown the MSB. This leads to the conclusion that the gasket was not drawn up into the drain line causing any blockage. Since the gasket material is not expected to have been drawn into the drain and fill line, it cannot impact the ability to reflood or fill the MSB during unloading operations because the MSB refloods through the drain line. No USQ, significant increases in occupational exposure, significant unreviewed environmental impact, nor changes in the license conditions of the C of C are created. (SE 99-017)

V. COMMITMENT CHANGE EVALUATIONS

1. Refueling Cavity. Our response to BUL 84-03 dated October 2, 1984 that states the cavity water seal and the keyway are physically inspected for signs of leakage requires a change in leakage identification method. Instead of a physical inspection, the containment Sump A can be monitored for seal leakage.

   Justification: Bulletin 84-03, "Refueling Cavity Water Seal," prompted an evaluation following a Haddam Neck failure of its inflatable reactor cavity seal. Past operating experience and safety measures in place have prompted a change in method for seal leakage identification. The change from inspection to monitoring the sump saves personnel radiation exposures. Measures are in place to ensure equipment malfunction does not occur should seal leakage or failure occur and the keyway fills with water. The water could be returned directly to the reactor vessel through Sump B and the core deluge lines. (CCE 99-005)

2. Refueling Outage Lessons Learned Review. A 1996 enforcement conference action that stated an interim review team would address the lessons learned during U2R22 and revise procedures as necessary requires clarification.

   Justification: The conference commitment included revisions to administrative procedures to ensure the commitment was met. This change addresses an implementation change that is an improvement to the method for returning systems to service. Instead of releasing control of selected plant systems at the beginning of an outage and regaining control at the end of an outage, real time tracking of work activities and issues are to be maintained. The SRO on watch will determine and accept system operability, not the system engineer as previously stated. (CCE 99-006)

3. Reactor Vessel Seal Inspection. A December 18, 1972 letter to the NRC stated that refueling procedure RP 1A does not have the incore thimbles withdrawn until after the reactor vessel cavity seal leakage inspection is complete. This inspection commitment is to be deleted.

   Justification: Operating experience has shown that an inspection of the reactor vessel seal is no longer required. Other seal leakage monitoring methods exist to replace the physical inspection such as measuring sump leakage for a quantifiable leakage determination. (CCE 99-007)

4. Primary-to-Secondary Leak Rate. Letter VPNPD-87-510 to the NRC dated November 20, 1987 that described our primary-to-secondary leakage monitoring program requires an update to include the adoption of EPRI TR-104788, "PWR Primary-to-Secondary Leak Guideline."

   Justification: The primary-to-secondary leakage monitoring program was designed to enable the monitoring of a tube failure with leakage characteristics similar to the North Anna event. The EPRI guidelines are improvements made since the 1987 commitment. A maximum leak rate of 150 gpd provides assurance against tube rupture at normal and faulted conditions and helps ensure that the dose contribution from tube leakage remains less than 10 CFR 100 dose limits for postulated faulted events. This is below the 500 gpd TS limit. (CCE 99-008)

5. Instrument Air Header Pressure. Our March 19, 1977 letter to the NRC that discussed our low alarm setpoint for instrument air header pressure needs to be rescinded.

   Justification: Modifications performed in 1981 included a redundant pressure channel to the pressurizer power-operated relief valve (PORV) controls. This provided activation capabilities during low pressure water solid conditions. The modification also installed nitrogen bottles to provide for PORV operability if required. Based on this facility change, the commitment to maintain the instrument air header pressure low alarm setpoint at 92 psig should be deleted. (CCE 99-009)

6. Duty and Call Superintendent. NUREG-0654 correspondence specified the requirement for having the position of duty and call superintendents. This requirement had been in our TS but has recently been removed; therefore, our commitments are to be updated.

   Justification: License amendments 190/195 were issued and state that the duty and call superintendent position is not necessary. From an emergency planning standpoint, senior plant management personnel will still fulfill the position of technical support center manager. This still provides an adequate level of staffing should an emergency situation arise. (CCE 99-010)

VI. NUMBER OF PERSONNEL AND TOTAL DOSE BY WORK GROUP AND JOB FUNCTION - 1999

Job Group Station Employees	Number of Personnel Greater Than 100 mrem	Total rem for Job Group	Work Function and Total Dose, rem					
			Reactor Operations & Surveillance	Routine Maintenance	Inspections	Special Maintenance	Waste Processing	Refueling
Operations	55	19.564	10.370	-------	5.773	-------	0.983	2.438
Maintenance	69	29.747	-------	10.027	3.625	------	2.473	13.622
Chemistry & Radiation Protection	33	19.384	16.402	-------	-------	-------	2.982	-------
Instrumentation & Control	14	2.671	-------	2.265	------	-------	-------	0.406
Administration, Regulatory Services & Licensing, QA, Engineering, and Training	20	8.315	5.795	-------	2.151	------	0.369	-------
Utility Employees	18	6.325	0.330	5.995	------	------	-------	-------
Contractor Workers & Others	294	108.373	3.755	11.990	10.133	82.222	0.273	-------
GRAND TOTALS	503	194.379	36.652	30.277	21.682	82.222	7.080	16.466

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

VII. STEAM GENERATOR INSERVICE INSPECTIONS

STEAM GENERATOR EDDY CURRENT TESTING

The following abbreviations are used throughout this report section.

xH	#x Tube Support Plate Hot Leg
xC	#x Tube Support Plate Cold Leg
AVx	Anti-Vibration Bar #x
TSH	Tubesheet Hot Leg
FBH	Flow Distribution Baffle Hot Leg
FBC	Flow Distribution Baffle Cold Leg
DNT	Dent (condition where the tubing inside diameter is less than nominal)
FSH	Free Span History (an MBM type indication that exhibited no change from the previous exam)
PLP	Possible loose part
PLP+VOL	Possible loose part plus volumetric indication

Unit 1

No eddy current testing was performed on Unit 1 in 1999.

Unit 2

Inspection Plan: Unit 2 steam generators were inspected from January 1, 1999 to January 8, 1999. Rows 3 and above were inspected full length using a bobbin coil (3394 tubes in each steam generator). Rows 1 and 2 were inspected over the straight length using a bobbin coil and a pluspoint was used for the U-bend (105 tubes in each steam generator). Additionally, the hot leg expansion transition area (-2", +3") in 20 percent of the tubes was inspected using a pluspoint (700 tubes in each steam generator). No tubes were previously plugged in either steam generator.

Inspection Results: The following table summarizes the number of tubes found with indications. Some tubes have more than one indication (numbers in parentheses are the total number of indications):

Indication	"A" SG	"B" SG
FSH	16 (32)	9
DNT	3	3
PLP	0	2
PLP+VOL	0	2

The following table lists each indication in the "A" steam generator:

Row	Column	Indication	Location	Inch Mark
2	5	FSH	FBH	18.25
28	21	FSH	06C	35.16
45	24	FSH	02H	36.52
74	31	FSH	01C	6.20
38	35	FSH	AV1	9.37
38	37	FSH	AV1	11.57
31	48	FSH	05H	23.73
31	48	FSH	05H	32.24
31	48	FSH	05H	38.61
31	48	FSH	05H	41.82
17	52	FSH	04H	40.70
17	52	FSH	06H	11.49
17	52	FSH	06C	28.20
17	52	FSH	06C	34.98
1	56	FSH	TSH	14.15
1	56	FSH	TSH	28.87
1	56	FSH	TSH	31.42
1	56	FSH	TSH	34.18
1	56	FSH	TSH	35.53
1	56	FSH	TSH	38.94
1	56	FSH	TSH	42.07
41	72	FSH	AV1	9.83
36	75	FSH	FBC	4.80
43	80	FSH	FBC	2.90
17	86	FSH	02H	40.31
18	93	FSH	O5C	12.25
15	94	FSH	04H	21.97
1	102	FSH	01C	6.03
1	102	FSH	01C	27.36
1	102	FSH	01C	32.30
1	102	FSH	FBC	15.01
1	102	FSH	FBC	17.94
21	4	DNT	AV1	4.89
13	28	DNT	07H	3.86
33	88	DNT	AV6	1.05

The following table lists each indication in the "B" steam generator:

Row	Column	Indication	Location	Inch Mark
23	4	FSH	01H	3.68
18	13	FSH	06C	15.41
33	36	FSH	06H	7.06
61	36	FSH	03C	33.71
60	55	FSH	04C	10.54
59	58	FSH	07H	12.43
45	62	FSH	AV5	13.11
41	64	FSH	01C	37.09
14	91	FSH	05H	36.76
18	21	DNT	AV6	18.12
13	66	DNT	07H	35.99
16	87	DNT	07H	6.64
19	46	PLP	TSH	0.09
55	70	PLP+VOL	TSH	0.03
57	70	PLP	TSH	0.07
56	71	PLP+VOL	TSH	0.07

Repaired or Plugged Tubes: No tubes were plugged in the "A" steam generator. Tube locations R55 C70 and R56 C71 were plugged in the "B" steam generator because of indication of a loose part on top of the tubesheet plus a volumetric indication. The tubes were conservatively plugged since no qualified sizing technique exists for small wear indications on the top of the tubesheet. An attempt was made to size the indications, for informational purposes, and the result showed that the wear indications were about 10% throughwall. The two tubes with just loose part indications in the "B" steam generator were left in service because there was no degradation of the tube, and the loose parts were removed from the steam generators during sludge lancing. Visual inspection in-bundle following sludge lancing confirmed that the items were removed.

VIII. REACTOR COOLANT SYSTEM RELIEF VALVE CHALLENGES

OVERPRESSURE PROTECTION DURING NORMAL PRESSURE AND TEMPERATURE OPERATION

There were no challenges to the Unit 1 or 2 reactor coolant system power-operated relief valves or safety valves during normal pressure and temperature operation in 1999.

OVERPRESSURE PROTECTION DURING LOW PRESSURE AND TEMPERATURE OPERATION

There were no challenges to the Unit 1 or 2 reactor coolant system power-operated relief valves or safety valves during low pressure and temperature operation in 1999.

VIV. REACTOR COOLANT ACTIVITY ANALYSIS

There were no indications in 1999 where reactor coolant activity exceeded that allowed by Technical Specifications in either Unit 1 or Unit 2.